NUREG-0528
Supplement No. 1

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

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

Our Safety Evaluation Report for the Zimmer Nuclear Power Station, Unit 1 (NUREG-0528), Docket No. 50-358, was issued January 1979. At the time that the report was issued, there were two outstanding issues listed in subsection 1.8 for which we had not reached our final position, 18 confirmatory issues listed in subsection 1.9 for which we were awaiting confirmation that the applicant had met our positions in a manner satisfactory to us and there was one item of disagreement listed in subsection 1.10 between the applicant and us. We stated that these matters would be resolved prior to a decision on issuing an operating license for the Zimmer Station. Since the issuance of NUREG-0528, the Advisory Committee on Reactor Safeguards considered the Zimmer operating license application at its 227th meeting and subsequently issued a favorable letter on March 13, 1979 to the Chairman of the Nuclear Regulatory Commission, stating some concerns which the Committee felt should be resolved to our satisfaction (Appendix B). This supplement to NUREG-0528 discusses the resolution of the issues mentioned above, other matters considered since the issuance of NUREG-0528 which we consider to be closed, and addresses the Advisory Committee on Reactor Safeguards concerns (Section 18 of this supplement). In addition, this supplement presents our bases for proceeding with licensing prior to resolution of the generic matters covered by our Task Action Plans outlined in NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants" (Appendix C). Appendix D to this supplement contains an errata to NUREG-0528. The sections and subsections of this supplement are numbered the same as the corresponding sections of NUREG-0528 which may be referenced for more detailed discussion of the issues and our positions.

We have conducted a thorough investigation of the March 28, 1979 incident at the Three Mile Island Power Plant, Unit 2. This investigation includes studies of the potential design deficiencies in the plant, plant operator response to the accident including operator errors and/or misinterpretation of plant instrumentation and all other aspects of the accident which might lead to information that would improve the safety of nuclear power plants. The results of our investigations which are documented in NUREG-0694, "TMI-Related Requirements for New Operating Licenses," and clarified in NUREG-0737, "Clarification of TMI Action Plan Requirements," have been applied to plants that are currently under construction and for which operating licenses have been applied, but not yet issued, such as the Zimmer Nuclear Power Station. New safety requirements arising from these investigations which are documented in NUREG-0694, "TMI-Related Requirements for New Operating Licenses," and clarified in NUREG-0737, "Clarification of TMI Action Plan Requirements," have been applied to the Zimmer Station to the extent that they are applicable. The results of this effort as applicable to the Zimmer Station are included in this supplement in Section 22.

Subsequent to the issuance of NUREG-0528, the staff raised a number of new safety issues which were not addressed in its original safety review. Resolutions of these issues are discussed in the appropriate subsections of this supplement.

The applicant is currently estimating completion of construction by November 1981.

1.5 Summary of Principal Review Matters

Subsection 1.5 of NUREG-0528 summarizes the principal matters considered in our technical review and evaluation of the information submitted by the applicant in the Zimmer operating license application. Although in the broad sense the safety matters arising from the staff's review of the Three Mile Island incident are covered by this summary, those specific matters stemming from TMI-2 and their resolution for the Zimmer Station are discussed separately in Section 22 to this supplement. In the case of seemingly conflicting information between the material in Section 22 and other subsections of NUREG-0528, Section 22 should be treated as the most current information.

1.6 Modifications to the Facility During the Course of Our Review

The modifications to the Zimmer Station resulting from the implementation of TMI-2 requirements are discussed in Section 22 to this supplement.

1.8 Summary of Outstanding Issues

The resolution of the outstanding issues listed in subsection 1.8 of NUREG-0528 are discussed in the appropriate subsections of this supplement.

Issues which remain outstanding for this supplement to NUREG-0528 are listed below with the appropriate subsection references. These issues will be resolved prior to a decision to issue an operating license.

Issue	Subsections
• Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment	3.10
• Environmental Design of Mechanical and Electrical Equipment	3.11
• Fire Protection	9.5

1.9 Summary of Confirmatory Items

The implementation status of the confirmatory items listed in subsection 1.9 of NUREG-0528 are discussed in the appropriate subsections of this supplement.

Issues which require confirmation are listed below with the appropriate subsection references. If implementation of some staff requirements may not be confirmed prior to a decision to issue an operating license, the issues remaining to be confirmed will be made a condition of the operating license.

Issue	Subsections
• Toxic Chemicals	2.2.1, 6.4.2
• Seismic Analysis	3.7.1, 3.7.2
• Design of Seismic Category I Structures and Systems	3.8.1, 3.8.2
• Category I Masonry Walls	3.8.2
• Fission Gas Release	4.2.3

Issue	Subsections
• Ballooning and Rupture, ECCS Cladding Model..............	4.2.3
• Seismic and LOCA Loading.................................	4.2.3
• Channel Box Deflection....................................	4.2.3
• Scram Discharge System....................................	4.6.2
• Feedwater and Control Rod Return Line Nozzle Cracking (Response to NUREG-0619)............................	5.2.1
• Stainless Steel Pipe Cracking (Response to NUREG-0313, Rev. 1)..	5.2.3
• Fracture Toughness (Response to NUREG-0577)..............	5.2.3
• Inservice Testing of Pumps and Valves....................	5.2.4
• Operator Action..	6.3.4
• BWR Scram System Pipe Break...............................	6.3.4
• Physical Separation of Associated Cables.................	7.1.3
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• Concern Regarding High Energy Line Breaks and Consequential Control System Failures..................	7.7.3
• Degraded Grid Voltage......................................	8.1.2
• Protection of Reactor Containment Electrical Penetrations..	8.1.2
• Station Blackout Events....................................	8.1.2
• Control of Heavy Loads.....................................	9.1.4
• Diesel Generators..	9.6.1
• Industrial Security Plan...................................	13.7

1.10 Items of Disagreement Between the Staff and Applicant

The one item of disagreement discussed in NUREG-0528 dealing with dewatering of compacted backfill under Category I structures has been resolved and is discussed in subsection 2.5.3 of this supplement.

1.11 NUREG-0737 "Clarification of TMI Action Plan Requirements

The staff has completed its review of the applicant's initial response to the TMI Action Plan requirements listed in NUREG-0737. These requirements have been approved by the Commission for near-term operating license issuance. 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 Informa.ion from Applicant	Confirmation by the Staff
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff................	X	
II.B.2	Plant Shielding to Provide Access.........	X	
II.B.3	Postaccident Sampling Capability...... ...	X	
II.B.7,8	Analysis of Hydrogen Control.............	X	
ıI.E.4.1	Dedicated Hydrogen Penetration............	X	
ıI.E.4.2	Containment Isolation Dependability.......	X	
II.F.1	Additional Accident-Monitoring Instrumentation Noble Gas Effluent Monitor.........	X	X
	Sampling and Analysis of Plant Effluents...............................	X	X
	Containment High-Range Radiation Monitor.................................	X	
	Containment Water Level Monitor...........	X	
II.F.2	Instrumentation for Detection of Inadequate Core Cooling................	X	
II.K.3	Final Recommendations of Bulletins and Orders Task Force		
	Item 13	X	X
	Item 15	X	
	Item 18	X	X
	Item 24	X	
	Item 27	X	
	Item 44	X	
III.A.1.1	Upgrade Emergency Preparedness..........	X	X
III.A.2	Improving Licensee Emergency Preparedness-Long Term..................	X	X
II.D.3.4	Cortro Room Habitability...............	X	X

2 SITE CHARACTERISTICS

2.2 Nearby Industrial, Transportation and Military Facilities

2.2.1 Transportation of Toxic Chemicals

In our safety evaluation (NUREG-0528) of nearby industrial, transportation and military facilities, we were able to conclude that the distances from the plant to potential carriers of hazardous cargo meet the guidelines described in Regulatory Guide 1.91, "Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites." However, we were not able to conclude that the guidelines described in Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," have been met in all cases because of the lack of information about toxic materials transported on nearby highways.

We stated in Section 6.4 of our Safety Evaluation Report of January 1979 that the control room is adequately protected against the hazards associated with the shipment of toxic gases along the C&O Railway and Ohio River near the site. We noted, however, that we were unable to conclude whether or not a hazard existed to the control room operators as a result of any accidental releases of toxic chemicals transported on U.S. Route 52 past the Zimmer site.

In March 1979, we requested additional information about toxic chemicals transported along U.S. Route 52 past the Zimmer site. Route 52 runs adjacent to the eastern border of the site. Guidance for means of obtaining sufficient and conclusive information was provided in staff position 312.33 (6.4). The requisite information described by the staff consisted of a three step screening process with regard to toxic materials that are normally shipped by truck. The first step involved the identification, in terms of toxicity and shipment size, of all chemicals which may pose a threat to the control room operators in the event of a release. The second step called for a determination of the shipping frequencies, so as to see if Regulatory Guide 1.78 guidelines were met. An alternate approach, in the form of a traffic survey on Route 52, was suggested in the event the shipment data could not be obtained by other means. Furthermore, it was noted that a risk assessment should be made for those chemicals which exceeded the Regulatory Guide 1.78 guidelines. In the third step, the need for protective measures was to be identified with respect to those chemicals which were determined to pose an undue risk.

The request for the above information stemmed from our finding that the average daily truck traffic on Route 52 past the Zimmer site is in the range of 400 to 600 trucks per day. Using an average value of 500 trucks per day, if the fraction of total shipments carrying toxic chemicals were to exceed about 0.0055%, then a detailed risk assessment should be made in accordance with Regulatory Guide 1.78.

In response to our concerns, the applicant examined the list of toxic chemicals listed in Regulatory Guide 1.78 and reviewed these on the basis of a 50,000 pound maximum truck shipment. The staff found this to be insufficient for an evaluation of the hazard since the toxic chemical listing provided in Reg. Guide 1.78 is illustrative, and it does not include many other toxic substances that may be transported past the site. Furthermore, the applicant's telephone and letter survey of the local manufacturers was not able to eliminate entirely the possibility that some of the potentially hazardous chemicals could be transported on Route 52.

In view of the above, the staff was not able to make a finding that the Zimmer site meets Regulatory Guide 1.78 guidelines and that the potential toxic chemical traffic past the site will not pose an undue risk to the control room operators. The staff has met with the applicant in April 1981 and discussed the means of resolving this open item. The applicant indicated that Route 52 did not have any weigh-station facilities and that it was not feasible to conduct a road traffic survey in order to assess directly the truck cargo composition. The applicant agreed, instead, to perform within the next several months, a detailed risk assessment with respect to the truck traffic on Route 52, taking into account statistical data such as truck accident rates, lading loss probabilities, and local meteorology. We believe this proposal to be reasonable and that it offers a means of resolving this question. We will review the applicant's assessment when it is received and report our evaluation and conclusion in a forthcoming supplement to this report.

Furthermore, we note that the applicant has provided control room isolation capability due to protection requirements against the potential effects of chlorine and ammonia. Hence, additional toxic gas protection, if found to be necessary, could easily be installed. In view of the above considerations, we believe that the concern with respect to toxic gas hazards from potential traffic accidents on Route 52 can be resolved satisfactorily prior to the operation of Zimmer Unit 1.

2.2.2 Quarry Operation

Since the issuance of NUREG-0528, the staff learned that a limestone quarry operation has been proposed for the Kentucky area across the Ohio River and a short distance from the Zimmer Station. The staff has reviewed this proposed operation for potential impact on the Zimmer operation resulting from the use of explosives.

Specifically the staff reviewed the blast effects (overpressure) on the Zimmer structures due to both routine blasting and accidental detonation of stored explosives on the earth's surface, and the potential for ground motion and free field loading at the Zimmer site.

At the present, there are approximately 41 tons of explosives stored 8700 ft. from the plant site. The staff's review considered "worst case" storage conditions which maximized expected ground effects and provided a factor of safety of approximately 3.

The staff concludes that the overpressure will be less than those of a design basis tornado for the Zimmer site and are therefore acceptable.

The staff also concludes that a free field peak ground acceleration of less than 0.02g and a vertical major principle stress in the soil of approximately 0.5psi due to blast overpressures are the maximum expected ground effects.

Because the expected effects are well below the design seismic and tornado loadings, the staff would not expect damage to site facilities as designed and thus considers the effects of possible mining operations acceptable.

For the purpose of this review two explosives storage modes were considered as they provide "worst case" assessment of blast and ground motion phenomena:

a. Surface Storage - accidental denotation of explosives will result in maximum air blast induced ground motions.

b. Subsurface Storage - accidental detonation of explosives will result in maximum energy partitioning to soil.

It is assumed that the quantity of explosives stored at the site will not exceed 1000 tons. At present the quantity stored is approximately 41 tons. Since pertinent cratering effects of an explosives detonation scale in approximate proportion to the cube root of the yield, the choice of a quantity of 1000 tons provides an effects factor of safety of approximately 3; i.e., overestimating the yield by a factor of 27 will result in a 3 fold overestimate of effects.

Surface Storage - For this mode it is assumed that the explosives are stored in standard protective structures at or near the surface. Upon accidental detonation of the entire 1000 tons, the staff has assumed the storage structures fail and the released energy is partitioned into the air and into the ground. That energy partitioned to the ground results in the formation of a crater; a zone of ruptured soil immediately beneath the crater; and a zone of plastically deformed soil immediately adjacent to the ruptured zone. At distances beyond the plastic zone transient earth motions will occur without permanent soil deformation of significance. In this regime, damage to structures at or near the surface of the earth due to direct ground shock is not expected as the magnitude of air induced ground shock is much greater and thus controls. The air induced ground shock results from the transfer of air blast energy into the soil at the ground surface and its magnitude is a function of the overpressure of the air blast wave immediately above it. The major principle stress in the soil will be essentially vertical and approximately equal to the air blast overpressure at the applicable distance and time. For a 1000-ton surface detonation at a distance of 8700 feet, the expected overpressure of the air blast front is less than 0.5psi.

For a 1000-ton surface detonation, the approximate magnitude of pertinent effects have been estimated to be: crater radius ≅65'; crater depth ≅30'; rupture zone radius (measured to extreme edge of the rupture area) ≅100'; plastic zone radius of significant deformation ≅200'; and vertical major principle stress at ground surface at a distance of 8700 feet from detonation point <0.5psi.

Conclusion - Surface Storage - As the location of the Zimmer plant is well outside the range of possible crater effects and the expected free field loading due to air blast overpressure is not significant considering the design seismic and tornado loadings, significant damage to site facilities is not expected from a surface storage detonation.

Subsurface Storage - For this mode it is assumed that the explosives are stored underground in a manner that would provide for maximum energy transfer between the explosives and the containing soil. To maximize the magnitude of expected ground motions at any distance it is also assumed that the explosives are stored in an enclosed area that will completely contain the detonation (no venting to the atmosphere) and that the storage area is small enough to preclude energy decoupling. Under these conditions, the free field pressure in the regime beyond the plastic zone would be solely due to direct induced ground shock. The peak acceleration of the ground at a distance R in miles from the detonation of 1000 tons of explosives can be estimated by the expression a Z0.06g/R2. Thus for a 1000 ton subsurface detonation located 8700 feet (1.6 miles) from the site one would expect to experience a peak ground acceleration of Z0.02g at the site.

Conclusion-Subsurface Storage - Since the maximum free field ground accelerations for the design operating basis earthquake are taken as 0.10g horizontal and 0.07g vertical, which are greater than the 0.02g estimated above, no significant damage is to be expected due to any possible underground detonation at the limestone mining site.

2.4 Hydrology

2.4.3 Water Supply

Siltation of the Intake Flume and Service Water Pump Structure (SWPS)

In April 1979 soundings were taken in the service water intake flume that indicated sediment accumulation varying from five to twelve feet. The sedimentation was identified due to service water pump cavitation and rotor failure during preoperational testing. Abnormal conditions during construction were determined to be at least partially responsible for this excessive sedimentation. The abnormal conditions were long duration flood stages on the river and no flow velocity in the flume during construction.

The service water pump structure consists of a concrete caisson sitting on rock. The intake flume is a steel sheetpile structure with the piles driven to rock. The intake flume is approximately 150 feet long and 30 feet wide and is angled downriver at 45° to prevent barge impact. The top of the sheetpile walls are at elevation 510 ft. msl. The flume has a concrete slab at the bottom at elevation 437 ft msl. A floating trash boom will be located at the entrance of the flume to prevent large floating objects from entering the flume. A bar grill is located at the entrance to the SWPS to prevent smaller objects from entering the pump bay.

The normal pool elevation of the Ohio River is 455 ft msl. The flood of record, which is the 1937 flood modified to reflect present conditions, is at elevation 508.5 ft msl. The Probable Maximum Flood, which is the design basis flood, is at elevation 549.4 ft msl.

The service water pump structure has four service water pumps, two on each side. The pump suction centerline elevation is 438.5 feet msl. The applicant has furnished detailed documentation of the potential sedimentation problems at the Zimmer site, including estimates of sedimentation rates for various combinations of plant water use rates and river flows. For average annual river flow, these estimates of sediment deposition range from 0.3 to 2.8 feet (depth) per month, and for the Probable Maximum Flood which would last about two months, the estimate is 7.3 feet. The staff has reviewed the sedimentation information and deposition estimates and made independent calculations. We conclude that the licensee estimates are conservative.

The applicant has installed five sediment monitors in the intake flume and SWPS. These monitors will be alarmed to sound in the control room on excessive sediment levels. The applicant has also developed procedures to periodically remove sediment from the flume. The plan calls for an intermediate station (i.e. a pump) located at the top of the intake flume wall which will pump silt slurry from the intake flume to the settling basin. The applicant has committed to have a backup pump available and stored in a location protected from the Probable Maximum Flood. The applicant has also committed to procedures for the relocation of the operating pump from the top of the intake flume for storage at higher elevations if the river should approach the 510 ft msl "top of flume" elevation. The applicant will not be required to remove silt when the river is above the 510 ft msl elevation.

The applicant has also installed a jetting system in the SWPS to preclude silt from accumulating. Additionally, the silt pump discussed above can also be used to remove silt from the SWPS.

Two critical scenarios, where possible blockage of the intake flume could interrupt the emergency water supply, have been investigated by the staff and are discussed below.

Ohio River Flooding

Sediment concentrations in the river increase with increasing river flows. The staff was concerned that during an Ohio River flood and concurrent failure of the downstream navigation dam, it might be possible to deposit enough sediment in the intake flume to completely block it. Following the postulated flood and dam failure, the river could recede to elevation 445.0 feet msl which would only allow for about 8.0 feet of sediment accumulation before blockage of the flume. For the PMF the licensee estimated about 7.3 feet of deposition during the duration (about two months or more) of the PMF. The staff made a very conservative estimate of about 10.0 feet of deposition during the PMF. The staff's proposed technical specification would require the licensee to initiate sediment removal when the level in the flume reaches elevation 441.5 feet msl. However, they would not be able to remove sediment from the intake flume when the river level is above elevation 510.0 feet msl, which is about 23 days during the PMF. This would leave the licensee more than a month for sediment removal before the river drops to a critical elevation. Based upon actual licensee tests of sediment removal rates and the amount of sediment that must be removed, the staff concludes that a flow path for river water can be maintained to the SWPS.

Ohio River Drought Conditions

In the event of a failure of the downstream dam during drought conditions, the minimum river level would be at elevation 442.0 feet msl. The proposed Technical Specification requires that sediment removal be initiated when it reaches elevation 441.5 feet msl. Additionally, during drought conditions sediment concentration in the river is very low. There would be little or no deposition in the intake flume. The proposed technical specification also requires that the plant be considered inoperable (requires shutdown) if the deposited sediment level reaches elevation 440.0 feet msl during drought conditions. With the plant shutdown, the reduced emergency water supply requirements could be obtained from other sources such as groundwater or portable pumps to the river. The staff concludes that this scenario will not endanger safety systems and further that siltation of the Zimmer intake can be controlled by the licensee's procedures and equipment.

The following is a proposed technical specification for siltation of the intake flume and SWPS.

Technical Specification - Sedimentation of Intake Flume and SWPS

Sediment buildup in the intake flume and SWPS can block the flow path for the emergency water supply. To preclude this situation from developing, a Limiting Condition for Operation will be established with regard to the accumulated level of sediment in the intake Flume and SWPS. The following limiting conditions for operation will apply at all times:

a. Clearing of Sediment From Intake Flume and SWPS. The sediment sensor system employed in the intake flume and SWPS will be set to sound an alarm in the control room when the sediment level reaches elevation 441.5 feet mean sea level (msl). With the alarm level exceeded in the flume or SWPS, the physical removal of deposited sediment shall be initiated within 24 hours and continued until all sediment is removed. The exception to this condition is that sediment removal does not have to be performed when the Ohio River level exceeds elevation 510.0 feet msl.

b. Upper Limit of Sediment Accumulation. When the sediment level reaches elevation 444.0 feet msl (as determined by the monitors or manual measurements) the plant will be considered inoperable.

c. Drought Conditions on the Ohio River. Drought conditions on the Ohio River near the Zimmer site are defined as the times when the Ohio River discharge is less than 10,000 cubic feet per second (cfs) as determined at the Markland Lock and Dam gage at river mile 531. During drought conditions on the Ohio River, the sediment level in the intake flume and SWPS shall be maintained below elevation 440.0 feet msl at all times. If the sediment level exceeds elevation 440.0 feet msl (as determined by the monitors or manual measurements) during drought conditions, the plant will be considered inoperable.

2.5 Geology, Seismology and Geotechnical Engineering

2.5.2 Seismology

Vibratory Ground Motion Summary of the 27 July 1980 Kentucky Earthquake

On July 27, 1980 at approximately 2.52 EDT an earthquake occurred near Sharpsburg, Kentucky, 77 kilometers from the Wm. Zimmer Nuclear Power Station. Because this earthquake was the largest event to occur in the eastern portion of the Central Stable Region (Michigan, Ohio and Kentucky) since 1937 the staff requested a thorough evaluation be completed by the applicant. Based on this information and staff review, we find no reason to change our original conclusions that 0.20g and 0.10g (along with a modified El Centro response spectrum) are adequate values for the Safe Shutdown Earthquake and the Operating Basis Earthquake, respectively. The staff also finds that the vibratory ground motion from the July 27, 1980 earthquake did not exceed that expected from the Operating Basis Earthquake and that postulated ground motion from an event similar to the July 27, 1980 earthquake would not have exceeded the Safe Shutdown Earthquake if such an event occurred near the site.

Seismological Parameters of the 27 July 1980 Kentucky Earthquake

Location of the 27 July 1980 Kentucky Earthquake

The National Earthquake Information Service (NEIS) has located the epicenter of the July 27, 1980 event at 38.174°N and 83.907°W. Other workers at the University of Michigan (UM) and the Tennessee Earthquake Information Center (TEIC) have located the epicenter in this same general area. These locations put the earthquake about 77 kilometers away from the Zimmer plant. Depth determinations of the earthquake were not well constrained, however an aftershock study suggests that the event occurred between 6 and 16 kilometers depth (Zollweg, 1980).

Magnitude of the 27 July 1980 Kentucky Earthquake

The National Earthquake Information Service has reported the magnitude of the July 27, 1980 earthquake to be m_b = 5.2 and m_{blg} = 5.0 to 5.3. Other investigators (Mauk and Christensen, 1980, Zollweg, 1980, Stevens, 1980) have calculated similar magnitudes (both m_b and m_{blg}) for the July 27, 1980 earthquake.

Intensity of the 27 July 1980 Kentucky Earthquake

Although various sources (such as the Universities of Kentucky and Michigan) have estimated the Modified Mercalli Intensity (MMI) for the July 27, 1980 earthquake the staff relies on the United States Geological Survey (USGS) to assign intensities. In the epicentral region (near the town of Sharpsburg, Kentucky) the USGS (Open File Report 81-198, 1981) has estimated the intensity to be MMI=VI while MMI=VII damage occurred about 50 kilometers to the north of the epicenter (Maysville, Kentucky). The closest town to the Zimmer site is Moscow, Ohio which has been assigned MMI=V by the USGS.

Intensity is a measure of observed damage and felt effects. The damage depends upon the size of the earthquake, its depth, the distance from the earthquake source, the nature of the geologic material between the source and the point of observation and the geologic conditions at the point of observation itself. Although an attempt is made in the intensity scale to account for differences in structural design, it is only done in a very general way. With this in mind it has been suggested (Anderson et. al., 1980, Mauk and Christensen 1980) that the higher intensities in Maysville are due to specific site conditions and the old age of the general brick masonry construction. The applicant has assumed in ground motion estimates that the epicentral intensity of this earthquake is MMI=VII. The staff concurs that this is a conservative and appropriate value (using Nuttli 1978, MMI=VII is associated with mblg = 5.25) for the epicentral intensity to be used in ground motion estimates of the July 27, 1980 Kentucky earthquake.

Ground Motion Estimates of the 27 July 1980 Earthquake at the Zimmer Site

As part of the Zimmer FSAR (Revision 70, Appendix K) the applicant has estimated the ground motion in the Zimmer site vicinity from the July 27, 1980 earthquake. As part of this analysis they have used different intensity attenuation relationships to predict the intensity observed at the site. Results show that at a distance of 77 kilometers using an epicentral intensity MMI=VII, intensities estimated vary between V and slightly less than VI (the intensity scale was treated as a continuous scale). The applicant states that the attenuation relationships used (Revision 70, Appendix K, pg. 20) tend to estimate an upper bound rather than a mean value of MMI. This results from the fact that isoseismals tend to enclose all reported locations of intensity except obviously anomalous sites. For the July 27, 1980 earthquake the staff weighs heavily the assigned intensity of the USGS for Moscow, Ohio and concurs with the applicant that MMI=V is an adequate estimate of the intensity at the Zimmer site.

Estimates of the peak ground motion at the site were completed by the applicant using magnitude (m_b = 5.1) epicentral intensity (MMI=VII) and site intensity (MII=V) (Revision 70, Appendix K, page 25). The most conservative estimate of peak acceleration is 0.07g while a more common value of 0.02g to 0.05g is predicted. Response spectra were estimated for the mean and 84th percentile levels of ground motion. Results show that the estimated ground motion from the July 27, 1980 earthquake was less than that expected from the OBE (peak acceleration 0.10g) at all frequencies. Based on staff review and the applicant's analysis we conclude that predicted ground motion from the July 27, 1980 earthquake to be less than expected from the OBE.

Ground Motion Estimates of an Assumed Earthquake at the Zimmer Site

The NRC staff also requested the applicant to estimate the ground motion at the Zimmer site assuming an earthquake similar to that which occurred on July 27, 1980 reoccurred near the site. One method of making such a comparison is to collect accelerograms from similar magnitude earthquakes, recorded at appropriate distances (less than 25 kilometers) and site conditions. Other methods involve estimating peak acceleration from source distance, magnitude and/or epicentral intensity. While all methods used have their merit the staff believes that site specific spectra results (when and where possible) are more realistic. Reasons for this include the belief that magnitude is a better estimator of source strength (then intensity) and actual records of ground motion recorded at distances less 25 kilometers do not include large attenuation effects (attenuation differences between the Eastern and Western United States).

For the site specific spectra comparison the applicant collected ten representative accelerograms (matching a magnitude of about 5.1 to 5.2 at an average distance of about 12 kilometers recorded at stiff soil sites). The SSE design spectrum (with a peak acceleration of 0.20g) envelopes all the site-specific response spectra that the applicant has used. Results of the empirical methods made by the applicant (using the epicentral distance of 15 kilometers, magnitude mb=5.1, and/or epicentral intensity, MMI=VII) also show that the SSE peak acceleration would not have been exceeded. The staff has compared the 84th percentile, m_b = 5.3 ± .5, soil site specific spectra done by Lawrence Livermore Labs (LLL draft Report, Seismic Hazard Analysis: Site Specific Response Spectra Results, Aug. 1979) to the SSE spectrum (modified El Centro) used by the applicant. The SSE design spectrum envelopes the 84th percentile of the LLL specific response spectrum for frequencies below 20 Hz. Based on staff analysis and review, and the applicant's analysis it is concluded that if an earthquake similar to that which occurred on July 27, 1980, reoccurred near the site, that the SSE design spectrum would not have been exceeded.

27 July Earthquake: Tectonic Significance

The July 27, 1980 earthquake occurred in an area which in the past has exhibited low levels of seismicity (Stover, Reagor, and Algermissen, 1979 Algermissen and Perkins, 1976). As required by 10 CFR Part 100 Appendix A, where epicenters or locations of highest intensity of historically reported earthquakes cannot be reasonably related to tectonic structure within the tectonic province of the site, the accelerations at the site shall be determined assuming that these earthquakes occur at the site. Because the July 27, 1980 earthquake occurred in close proximity to known surface faults (West Hickman Creek Fault Zone, Zimmer FSAR, Appendix K) there may be some suggestion that the event occurred on one of these faults. Various investigators (Herrmann, 1980; Mauk, Christensen and Henry 1981; Zollweg 1980) who are studying this earthquake have determined focal mechanism solutions with right-lateral strike slip motion with a thrust component. This data supplemented by an aftershock study (Zollweg 1980) suggest that the fault plane trends to the northeast and dips to the southeast. Although this has been one of the better studied mainshock aftershock sequences in the Central United States, the staff cautions against attaching the July 27, 1980 earthquake to a known surface fault. Depth estimates of the mainshock and aftershocks range from 6 to 16 kilometers (Zollweg 1980; Mauk, Christensen and Henry 1981). Little if any information is known about specific surface fault projecting to these depths.

With this in mind the staff concludes that earthquakes similar to the July 27, 1980 Kentucky earthquake (mb=5.2, mblg=5.0 to 5.3 MMI=VII) along with the March 9, 1937 Anna, Ohio earthquake (mb=5.0 to 5.3, MMI=VII-VIII) could occur anywhere within the Central Stable Region of the United States, and that this practice is conservative. As more work is completed on earthquakes in the Central United States a better understanding of seismic source structures along with limiting the locations of earthquakes of this size is anticipated.

Conclusion

In determining the Safe Shutdown Earthquake, and the Operating Basis Earthquake the tectonic province approach described in Appendix A of 10 CFR Part 100 was followed. The applicant's proposed Safe Shutdown Earthquake acceleration level of 0 20g (along with a modified El Centro response spectrum) and Operating Basis Earthquake Acceleration level of 0.10g are conservative.

The staff also finds that the vibratory ground motion from the July 27, 1980 earthquake did not exceed that expected from the Operating Basis Earthquake and that postulated ground motion from an event similar to the July 27, 1980 earthquake would not have exceeded the Safe Shutdown Earthquake if such an event occurred near the site.

2.5.3 Stability of Subsurface Material and Foundations

Foundation Preparation

In NUREG-0528, we stated, "It is our position that the compacted backfill material within the clay envelope be dewatered during plant operation, when necessary as intended at the construction permit stage and as described in Section 2.5.4.5.1.3 and Fig. 2.5-49 of the Final Safety Analysis Report. Such measures will assure stability of the foundation backfill in the event liquefaction occurs at the site, provide a means for collecting and discharging infiltrated water which may be trapped in the encapsulated fill, assure adequate effective stresses between foundations and fill, and provide a means for determining the effectiveness of the clay blanket and for detecting anomalous conditions. Water levels in the encapsulated backfill shall be maintained at or below elevation 457 feet above mean sea level measured at the backfill dewatering well. The applicant agreed to implement this position in accordance with the construction permit stage commitment but has taken issue with the dewatering level of 457 feet above mean sea level. The applicant believes that the construction permit stage commitment was 480 feet above mean sea level and that 480 feet provides adequate protection against excessive pore pressure in the compacted backfill. We will continue discussion with the applicant and try to resolve this detail prior to reactor operations. Until the matter is resolved, we will require maintenance of the 457 foot level during operation. The applicant provided us with the description and location of the dewatering system. We find the applicant's commitment and implementation acceptable provided agreement is reached on the dewatering level to be maintained. The resolution of this matter will be provided in a supplement to this report and the required dewatering level will be specified in the technical specifications."

The applicant committed to initiate pumping at a water level of 460 feet above mean sea level and to cease pumping at 458 feet above mean sea level. We find this commitment satisfactory since maintaining water levels below 460 feet mean sea level provides adequate assurance of foundation backfill stability. We consider this matter to be closed.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.2 Classification of Structures, Components and Systems

3.2.2 System Quality Group Classification

The staff has re-reviewed the applicant's Q-List contained in the Final Safety Analysis Report for completeness. The results of this re-review are discussed in subsection 17.2 of this supplement.

3.5 Missile Protection

3.5.2 Turbine Missiles

By letter, dated September 10, 1980, the staff requested additional information with regard to the Zimmer low pressure turbine disc. This information was needed in order for the staff to complete its review of the inservice inspection program. The applicant responded to the request by letter dated December 1, 1980. The staff evaluation is as follows:

Low-Pressure Turbine Disc Inspection

During November 1979, the NRC became aware of a problem of stress corrosion cracking in the Westinghouse turbines. Meetings were held with Westinghouse to ascertain the probable extent and severity of the problem. Westinghouse was recommending early inspection of turbines that had long operating times, and particularly those machines with discs of marginal material properties and history of secondary water or steam chemistry problems. Since then, inspections have been performed on about 25 more operating Westinghouse turbines, with indications of cracking, some severe, found in most of them. Investigations are continuing.

The applicant has submitted to the staff the material properties of the low-pressure turbine discs, as well as the calculations of critical crack sizes. The method used by the applicant to predict crack growth rates is based on evaluating all the cracks found to date in Westinghouse turbines, past history of similar turbine disc cracking, and results of laboratory tests. This prediction method takes into account two main parameters; the yield strength of the disc and the temperature of the disc at the bore area where the cracks of concern are occurring. The higher the yield strength of the material and the higher the temperature, the faster the crack growth rate will be.

We have evaluated the data submitted by the applicant and, in addition, performed our own calculations for crack growth and critical crack size. We conclude that Zimmer Unit 1 may be safely operated until the first refueling outage, at which time the LP turbine discs should be inspected. (See subsection 10.2 of this supplement.)

3.7 Seismic Design

3.7.1 Seismic Input

3.7.2 Seismic System and Subsystem Analysis

Seismic Analysis

Soil-Structure Interaction Analysis

Originally on the soil-structure interaction analysis the applicant used a two-dimensional finite element shear-beam model to characterize the soil under the plant structures. However, recent developments have shown that markedly varying results can be obtained by various methods of soil-structure interaction analyses. In view of this fact we requested the applicant to check the results of the original design analyses using the elastic half-space modeling technique (frequency independent compliance functions) employing current Regulatory Guide 1.60 and 1.61 Criteria for the Safe Shutdown Earthquake.

The applicant has submitted a comparative study of results obtained by two approaches to soil-structure interaction analyses, namely, the finite element modeling and the layered viscoelastic half-space modeling techniques. We have reviewed the analytical procedure and theoretical basis (Luco, J. E. "Vibrations of a Rigid Disc on a Layered Viscoelastic Medium," Nuclear Engineering and Design 36, 1976) employed by the applicant for the viscoelastic half-space approach and found that the method as implemented is acceptable.

The theoretical development of the methodology is consistent with the assumptions therein. However we have found that the soil damping values used in the analysis were appreciably higher than expected and the results of the analyses may not be representative of the actual condition. Therefore, we have requested the applicant to make re-analyses using lower soil damping values, and the applicant has committed to do so.

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

3.8.2 Concrete and Structural Steel Internal Structures

The original Design Assessment Report (DAR) was submitted in July, 1979 and was reviewed, commented, and accepted by the staff with the fluid-structure interaction analysis being an outstanding issue. Since then, the applicant has considered the fluid-structure interaction effects resulting from pool dynamics events in the design of the containment and its interior structures.

In late 1980 the applicant submitted a revised DAR which incorporated all pertinent information from the original DAR plus amendments 1 through 12, the Closure Report, and all the information generated since the submittal of the Closure Report. We have reviewed the sections of the revised DAR related to structural engineering and found that the loads and loading combinations, the methods of analysis and the acceptance criteria are basically the same as those contained in the original DAR as amended as the result of our previous review and comments. However, the applicant has not completed the assessments for structures such as reactor support, drywell floor and drywell floor columns. Furthermore, the results of the reassessment for the containment sturcture indicate that the margin factors in some portions of the containment structure are less than one. In view of these findings we require the applicant to complete the assessment of all structures and to reassess the overstressed portions of the containment structure using more realistic loads.

Concrete Containment Leak Chase Channel

The original purpose of the leak channels was for leak testing of the liner seam welds under pressure. All liner seams are covered with structural steel channels generally of size C3 x 4.1 welded continuously to the liner with a 3/16-inch fillet weld on each side of the channels. The channels as well as the liner are not considered as structural elements.

In the process of construction inspection by NRC regional personnel, some small safety-related instrument lines and conduits have been found to be attached to the leak chase channels. Questions were raised on the appropriateness of attaching the safety-related items to the non-structural leak chase channels, and on the integrity of the liner itself as a result of such attachments.

In response to the staff's concerns the applicant had its architect/engineer. Sargent and Lundy perform a study on the adequacy and effects of such attachments. The results of the study are contained in Sargent and Lundy Report No. SAD-348 and in the responses to the staff's review questions on the report. The staff has reviewed the report and the responses and found that the assumptions used in the analysis are conservative and the loads and load combinations used are comparable to those used in the design of seismic Category I steel structures. The stresses and strains as calculated are within the ASME Code, Section III, Division 2 Limits for the liner and its anchors. In the report it is indicated that the leak chase channel and weld material and welding QA/QC procedures are the same as those for the liner.

The leak chase channel welds have also been subjected to Vacuum Box Test during the 52 psi pressure test of the channel to assure leak-tightness of the liner seam welds and leak chase channel welds.

On the basis of the results of the study it is concluded by the applicant that the leak chase channels and their welds are structurally adequate to support the loads imposed through the attachments and that the liner and its anchor system are capable of taking the loads transmitting through the leak chase channels.

The staff concurs with this conclusion. However, in order not to encroach further on the margin of safety of the liner and its anchor system, no more such attachments should be made without NRC prior approval.

Masonry Wall

In addition to concrete and steel structures, there are also concrete masonry block walls in seismic Category I structures, including the reactor building. These walls are basically non-load bearing walls. They are mostly used as curtain walls and filler walls. That is, they are not used as structural elements to support any load other than their own weight. However, there are conditions in which concrete masonry walls are used to support seismic Category I components such as pipes. There are walls located near to safety-related components, equipment or systems. Therefore, the failure of these walls will jeopardize the functionality of these safety-related items. These concerns were transmitted to the applicant in April, 1980. The applicant provided information (by a letter, dated July 18, 1980) on masonry walls in response to our concerns. Subsequent to our review of the applicant's response, additional information was requested and provided by the applicant in a letter dated November 19, 1980.

For evaluation and design of the masonry walls, the applicant is using the criteria specified by the National Concrete Masonry Association, "Specification for the Design and Construction of Load-Bearing Concrete Masonry," April 1974. These criteria are being evaluated against our interim masonry wall evaluation criteria. The applicant will be requested to justify and resolve the differences between its criteria and our interim criteria.

Based on the findings of our preliminary review of the applicant's submittals, the staff does not agree with the rationale used by the applicant in his design/ analysis of the masonry walls in the following specific areas:

(a) It is unconservative to multiply the National Concrete Masonry Association (NCMA) allowable stresses by 1.67 since the NCMA values are high in comparison to UBC allowable stresses initially.

(b) The use of allowable stresses for type "M" mortar where type "N" mortar was actually used is not justified, regardless of test results which show in-place mortar strength stronger than type "M" mortar. If type "M" mortar were actually used, the in-place strength would be that much higher.

(c) Masonry walls will physically have some restraint at the wall ends, and therefore, the assumption that thermal loads may be neglected because pinned-end conditions were assumed is unconservative. It is necessary that thermal loads be considered by making the appropriate assumptions.

(d) With regard to local block shear pull-out loads at all attachments, only mortar joints around the concrete block or blocks to which the load is directly attached should be considered as the boundary for shear transfer.

(e) Where masonry walls are set on concrete curbs, shear friction alone should not be relied upon to keep the masonry walls from sliding off of the curbs. Positive mechanical means must be provided to attach the masonry walls to the concrete curbs.

(f) Where calculated stresses exceed allowable values, calculations of the wall geometric properties should be based on a "cracked" section.

The applicant has committed to resolve the masonry wall issues to the satisfaction of the staff either by justifying the differences between staff's interim criteria and the criteria used in Zimmer or by strengthening the walls as required before the fuel loading date.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System Analysis and Testing

Seismic Qualification of Mechanical Equipment

In NUREG-0528 we stated that "The applicant submitted procedures for dynamic testing and analysis techniques to confirm the adequacy of seismic Category I mechanical equipment, including their supports, to function during and after an earthquake of magnitude up to and including the safe shutdown earthquake.

"In instances where components have been qualified by testing to other than current standards such as Institute of Electrical and Electronics Engineers Standards 344-75, such components, particularly those vital to the actuation and continued operation during and after an earthquake of magnitude up to and including the safe shutdown earthquake, may have to be retested. Our seismic qualification review team is reviewing the nuclear steam supply system and balance-of-plant equipment lists and has inspected the Zimmer Station balance of plant equipment already installed at the site. This review will evaluate the qualification testing to determine that the effects of the combination of seismic and hydrodynamic loads have been properly accounted for. On the basis of the review audit and site visit, the seismic qualification review team will ascertain whether any nuclear steam supply system or balance of plant equipment components have to be retested. We initiated discussions with the applicant to develop a mutually acceptable resolution of any problems arising in this area. We expect a timely resolution of this issue and will present the results in a supplement to this report."

The status of this matter is discussed in subsection 3.10 below.

Dynamic System Analysis and Testing Piping Vibration Operational Test Program

The applicant has agreed to perform a piping preoperational vibration dynamic effects test program to check the vibration performance of piping important to safety. It is the staffs position that all essential safety related instrumentation lines should be included in this piping preoperational vibration test program. The applicant has agreed to include all safety related small bore piping and instrumentation lines as we requested.

In addition, the applicant has committed to perform pre-service inspection and pre-operational testing of all safety related snubbers as requested and to document these results as part of the piping pre-operational vibration test program.

3.9.2 American Society of Mechanial Engineers Code Class 2 and 3 Components Design, Load Combinations and Stress Limits

Load Combinations

In Section 3.9.2 of NUREG-0528, we stated that the applicant had agreed to reassess the structural margin of those ASME Code Class 1, 2, and 3 items which were designed by combining the peak dynamic responses to dynamic loads by the square-root-of-the-sum-of-the-square (SRSS) method. This reassessment was to use the absolute sum method of combining these responses.

Since that time our review of the SRSS method has continued. The use of the SRSS method for combining peak dynamic responses due to the loss-of-coolant accident and safe shutdown earthquake has been accepted by us in NUREG-0484, Rev. 1, "Methodology for Combining Dynamic Responses." The use of the SRSS method for combining dynamic responses to other loads in BWR Mark II plants has been approved by us under certain conditions in a letter from J. R. Miller (NRC) to G. G. Sherwood (GE), dated 6/19/80, concerning our review of the GE topical report NEDE-24010-P, "Technical Bases for the Use of the Square Root of the Sum of the Squares (SRSS) Method for Combining Dynamic Loads for Mark II Plants." We have determined that the Zimmer plant meets the conditions set by this letter. Therefore, the use of the SRSS technique for Zimmer is acceptable, and the applicant is not required to continue its absolute sum reassessment.

Response of a pipe run under seismic or supression pool swelling loads may be considered as a resultant of two component responses: the response due to inertia effects of its own mass and the response due to relative movement of its anchors. These two component responses generally have different dominant frequencies and have their individual maximum response occurring at different times. Since the response due to inertia effects is calculated using spectrum analysis and the response due to relative anchor movement is calculated using static analysis, only the values of individual maximum response are calculated. For obtaining maximum combined response, the applicant has used the SRSS method. Since these two responses are dynamic in nature and have random time phasing between their individual maximum responses, the staff concludes that the use of SRSS is acceptable due to the expected nonexceedance probability of the combined response using SRSS being greater than our criteria threshold in NUREG-0484, Rev. 1. Our findings are as follows.

The specified design and service combinations of loadings as applied to ASME Code Class 1, 2, and 3 pressure retaining components in systems designed to meet seismic Category I standards are such as to provide assurance that, in the event of an earthquake affecting the site or other service loadings due to postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity. The design and load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components comply with Standard Review Plan Section 3.9.3 and satisfy the applicable portions of General Design Criteria 1, 2, and 4.

Containment Pool Dynamic Loads

With regard to the issue of containment pool dynamic loads, the applicant has agreed to meet the staff acceptance criteria and has completed a Class 1 fatigue analysis of the ASME Class 2 & 3 downcomers and safety relief valve discharge piping using the ASME Code Class 1 fatigue rules. We have reviewed the results of this fatigue analysis and find them acceptable.

The applicant has calculated loads based on the NRC approved criteria for the Zimmer plant.

The NRC accepted loads and resulting stresses for the load combinations required by us are to be documented in the FSAR. Subsequent to the final determination of acceptable values for the Zimmer plant hydrodynamic loads, the applicant will reconcile the loads used for the plant design with the final accepted loads. Provided the final load values do not exceed the values used in the plant design, we consider this approach acceptable.

Pump and Valve Operability Assurance Program

The pump and valve operability program has been found acceptable subject to the satisfactory completion of the NRR Equipment Qualification Branch (EQB) site inspection and audit. Based on the EQB findings, there may be a need for retesting of selected vital appurtenances to pumps and valves. On satisfactory resolution of any outstanding items based on the EQB findings, we will consider this issue resolved.

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

On January 13, 1981, we issued a request for information to the applicant, which asked for information concerning their equipment qualification for seismic and hydrodynamic loads. The applicant responded to this request for information on January 20, 1981. Subsequently a request for additional information was issued on March 16, 1981. The applicant completed its response to this request for information on April 22, 1981. Based on these responses, the Seismic Qualification Review Team (SQRT) plans to conduct a plant site review of the applicant's qualification documentation during the week of June 1, 1981. We will report on the results of our review in a supplement to this report.

3.11 Environmental Design of Mechanical and Electrical Equipment

3.11.1 Discussion

By letter, dated November 9, 1979, the staff requested that the applicant conduct an environmental qualification program in accordance with current requirements. The applicant responded to the November 9, 1979 letter on April 10, 1980 with a schedule for completing the program. Several letters of guidance have been issued subsequent to the November 9, 1979 letter. In particular an October 1, 1980 letter requested a status report by November 1, 1980. The applicant issued the status report on January 7, 1981.

The staff's review of this matter is in progress and a site audit has been scheduled for the summer of 1981. We will confirm the environmental qualification of essential mechanical and electrical equipment in a future supplement to this report.

4 REACTOR

4.2 Fuel System Design

The objectives of the fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channeling to permit removal of residual heat even after a severe accident.

To meet the above-stated objectives of the fuel system review, the following specific areas are criticality examined: (a) design bases, (b) description and design drawings, (c) design evaluation, and (d) testing, inspection, and surveillance plans. In assessing the adequacy of the design, several factors involving operating experience, prototype testing, and analytical predictions are assessed in terms of specific acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability. The acceptance criteria are provided in Section 4.2 of the NRC Standard Review Plan (Ref. 1). Upon review and approval, the fuel system design limits provided in the applicant's FSAR become, for normal operation (including anticipated operational occurrences), the "specified acceptable fuel design limits" (SAFDLS) referred to in General Design Criterion 10 (Ref. 2).

4.2.1 Design Bases

The Zimmer FSAR separates the fuel system design bases into two main categories of discussion: (1) "general" design bases and (2) "detailed" design bases. Under the "general" category, the design bases are subdivided in terms of (a) "safety" and (b) "power generation." The safety design bases are presented in terms of preventing fuel damage that would result in significant releases of radioactive fission products from the fuel.

In regard to the "power generation" design bases, the fuel assembly design is defined (in conjunction with the core nuclear, thermal, and hydraulic characteristics and the characteristics of the instrumentation and protection system) to ensure that fuel change limits will not be exceeded during planned normal operation of "normal operational transients." The "detailed" design basis considerations are more detailed and include material selection factors, the effects of irradiation, fuel rod dimensional changes, incipient UO_2 center melting, the fuel rod's capacity for fission gas inventory, maximum allowable stresses, maximum internal gas pressure, flow-induced fuel rod vibrations, operational fuel rod deflections, fretting corrosion, internal pressure and cladding stresses during normal operations, cycling and fatigue limits, shock and seismic loadings, potential for water-logging rupture, potential for hydriding and other effects.

4.2.2 Description and Design Drawings

The fuel design described in the Zimmer FSAR is identical, except for channel thickness, to the General Electric 8x8 fuel design currently in operation in several BWRs and described in GESSAR (Ref. 3) and the generic reload report (Ref. 4). The channel box thickness in the Zimmer fuel design is 0.020 inches thicker than in standard 8x8 assemblies. The Zimmer fuel design channel box thickness is identical to that described in GE reports on BWR/4-5 fuel design (Ref. 5) and BWR/6 fuel design (Ref. 6) but the Zimmer fuel rods will not contain the natural uranium fuel pellets used in the 8x8 I (I = "improved") design described in those reports.

Mechanical and operating parameters for the 8x8 fuel assemblies are compared in Table 4-2 with the previously used (older design) 7x7 BWR fuel assemblies. The smaller diameter 8x8 rods, with lower linear heat generation rates and increased cladding thickness-to-diameter ratio, are intended to increase safety margins with respect to maximum linear power and fuel temperatures. In addition, the Zimmer 8x8 fuel assemblies have the following features not found in the (originally) standard 7x7 rods: (1) finger springs for controlling moderator/coolant bypass flow at the interface of the channel and lower tie plate, and (2) bypass flow holes drilled in the lower tie plate to provide an alternate flow path.

Table 4-2 Comparison of parameters for 7x7 and 8x8 fuel assembly designs

Parameter	7x7	ZIMMER 8x8
Fuel Rods/Assembly	49	63
Channel Thickness (in.)	0.080	0.100
Active Fuel Length (in.)	144	146
Uranium Weight/Assy. (lbs.)	412.8	409.7
Rod-to-Rod Pitch (in.)	0.738	0.640
Water/Fuel Ratio (cold)	2.53	2.60
Cladding OD (in.)	0.563	0.493
Cladding Thickness (in.)	0.037	0.034
Thickness/Diameter Ratio	0.0657	0.0689
Fuel Pellet OD (in.)	0.477	0.416
Pellet/Clad Diametral Gap (mils)	12	9
Maximum Linear Heat Generation Rate (kW/ft)	17.5	13.4
Maximum Fuel Temp. (F)	4380	3325

4.2.3 Design Evaluation

As part of our fuel system design evaluation, we review the applicant's methods of demonstrating that the design bases are met. Those methods include the results of prototype testing, analytical predictions, and operating experience, which for GE BWR fuel, has, in general, been very good. However, over the years some fuel damage mechanisms have been encountered and have required remedial action. The nature of those damage phenomena and the manner in which they will be accomodated by the design, operation, or surveillance of Zimmer fuel are described below.

Fuel Densification

One operating experience phenomenon that was first encountered in LWR fuels in the early 1970s, and which can affect the thermal performance of the fuel, involves the inreactor densification of the UO_2 fuel pellets. Briefly stated, inreactor densification (shrinkage) of oxide fuel pellets (a) may reduce gap conductance, and hence increase fuel temperatures, because of a decrease in pellet diameter; (b) increases the linear heat generation rate because of the decrease in pellet length; and (c) may result in gaps in the fuel column as a result of pellet length decreases--these gaps produce local power spikes and the potential for cladding creep collapse. The favorable results of our review of General Electric densification methods and other general information on fuel densification can be found in NUREG-0085 (Ref. 7).

Fuel performance calculations that account for some specific effects of fuel densification have been performed with an approved version of the General Electric analytical model, GEGAP-III (Refs. 8, 9). The approved analytical model incorporates time-dependent fuel densification, time-dependent gap closure, and cladding creep-down for the calculation of gap conductance.

Other fuel performance predictions, such as cladding mechanical response, are calculated with the General Electric integral fuel design models (Ref. 10). Cladding collapse has not been observed in BWR fuel rods, but its theoretical occurrence is calculated with an approved code, SAFE-COLAPS (Ref. 11), at core residence times in excess of 5 years, which is greater than the lifetime of the fuel.

Fission Gas Release

In 1976 we questioned (Ref. 12) the validity of fission gas release calculations in most fuel performance codes including GEGAP-III for burnups greater than 20,000 MWd/tU. GE was informed of this concern and was provided with a method (Ref. 13) of correcting gas release calculations for burnups greater than 20,000 MWd/tU. Although a reanalysis has not been specifically performed for the Zimmer fuel, an 8x8 reanalysis (Ref. 14) performed for early reflooding plants reportedly bounds the Zimmer case. In the generic reanalysis, fuel rod internal pressure was shown to remain below system pressure for rod peak burnups below 40,000 MWd/t. This conclusion remains unchanged for the newer prepressurized fuel design as well (Ref. 15).

The generic reanalysis did, however, result in higher initial stored energy and rupture pressure in the loss-of-coolant accident (LOCA) analysis. Under LOCA conditions, the higher fission gas release results in a maximum increase of 85°F in calculated peak cladding temperature (PCT) at end-of-life (~<33,000 MWd/tU planar average exposure). This added temperature increment results in calculated peak cladding temperatures of less than 2100°F for average burnups below ~ 33,000 MWd/tU and, thus, would not violate the 2200°F LOCA PCT limit. Nevertheless, this increase in PCT has not been accounted for explicitly in the Zimmer ECCS analysis. Therefore, fission gas release effects are found to be adequately analyzed for early life operation, but must be fully reanalyzed prior to exceeding a peak local burnup of 20,000 MWd/tU. This reanalysis may be performed with GEGAP-III modified to include the NRC correction method unless another approved code is available at that time. It is noted that a new GE code, which accounts for the burnup variable, is at an advanced stage of review and should be available for the reanalysis. This matter will become a condition of the operating license.

Ballooning and Rupture

In another LOCA-related area of concern, we have been generically evaluating three fuel material models that are used in emergency core cooling system (ECCS) evaluations. Those models predict cladding rupture temperature, cladding burst strain, and fuel assembly flow blockage. We have (a) discussed our evaluation with vendors and other industry representatives (Ref. 16), (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis" (Ref. 17), and (c) required licensees to confirm that their operating reactors would continue to be in conformance with 10 CFR 50.46 if the NUREG-0630 models were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed (Refs. 18 and 19).

Until we have completed our generic review and implemented new acceptance criteria for cladding models, we will require that the ECCS analyses in the Final Safety Analysis Report be accompanied by supplemental calculations to be performed with the materials models of NUREG-0630. For these supplemental calculations only, we will accept other compensatory model changes that may not yet be approved by the NRC, but are consistent with the changes allowed for the confirmatory operating reactor calculations mentioned above. We will report on the resolution of this issue in a supplement to this report.

Seismic and LOCA Loadings

Analytical results for the fuel assembly response to the combined effects of an earthquake and a LOCA have been provided in the form of a reference to a GE report, "BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," NEDE-21175-P, dated November 1976 (Ref. 20). Those results were said to apply also to BWR/4 and BWR/5 fuel assemblies.

Our review of NEDE-21175-P is now finished (Ref. 21). In May 1979 we found the analytical methods acceptable, but the generic fuel assembly design limits in that report were not accepted because the NRC had not yet completed development of the necessary acceptance criteria (this issue was then part of an NRC Unresolved Safety Issue, A-2 in . 22). Those acceptance criteria have now been completed, however, and are given in Ref. 23.

To fully resolve this issue, the applicant must supply results obtained with the approved GE analytical methods that show compliance with the acceptance criteria mentioned above. We will report on the resolution of this confirmatory issue in a supplement to this safety evaluation.

Gadolinia Poison

Several U-235 enrichments are used within each fuel assembly to reduce the local power-peaking factor. Gadolinium, a burnable poison, is also used to supplement the enrichment pattern and control rods in flattening the power distribution of the core. The gadolinium is in the form of gadolinia-urania pellets and is used in some of the interior rods in a variable axial concentration distribution. Gadolinium-bearing fuel was first incorporated as a regular component into the initial cores of Quad Cities Units 1 and 2, with operation starting in 1971 and 1972, respectively. Since 1965, a substantial number of test and regular gadolina-urania rods have been successfully irradiated to appreciable exposures. That operating experience, plus the physical and irradiation characteristics and material properties of GE gadolinia-urania fuel, are discussed in a GE topical report (Ref. 24) that is under review. Following review and acceptance of that report, it will serve as a referential document on the use of urania-gadolinia in GE BWR fuel assemblies; i.e., the report may be incorporated in plant safety analysis reports by reference without the need for further review. Should the ongoing review indicate that further action is needed regarding the use of this material in Zimmer Unit 1, we would notify the operators and require compliance with those needs. In view of the successful operation to date of gadolinia-bearing fuel rods, no such action is currently anticipated.

Waterlogging

The potential and consequences of operating with waterlogged fuel rods was addressed in the Zimmer FSAR, and we have independently reviewed the safety aspects of waterlogging failures generically (Ref. 25). The term "waterlogging" refers to the presence of water in the interior of a fuel element that has a prior cladding breech. As the fuel temperature increases rapidly during a power ascension, the steam formed inside the defected fuel rod might not escape through the defect opening fast enough to relieve the internal steam pressure; thus further rupture or splitting of the cladding could conceivably result. However, our survey (referenced above) of the available information, which includes (1) test results from SPERT and NSRR in Japan and (2) observations of waterlogging failures in commercial reactors, indicates that rupture of a waterlogged fuel rod should not result in failure propagation or significant fuel assembly damage of sufficient magnitude to affect the coolability of the fuel rod assembly. Failure propagation due to waterlogging is being investigated further as a part of a generic fuel failure propagation study underway at the Los Alamos Scientific Laboratory (Ref. 26). Results of that study have so far provided further support for the view that waterlogging failure will not lead to fuel failure propagation during normal reactor operation. While off-normal conditions are under further study, we conclude that based on (a) the operating experience with waterlogged fuel and (b) test reactor results, the evidence currently available provides reasonable assurance that the occurrence of waterlogging failures would not pose a threat with regard to either failure propagation or fuel coolability.

Pellet/Cladding Interaction

Another failure phenomenon that has been encountered in operating BWR (and PWR) fuel is pellet/cladding interaction (PCI). PCI generally occurs during a power increase as the fuel pellet expands and exerts stresses on the cladding. Although the exact mechanisms that contribute to PCI damage have not been established beyond a doubt, operating experience indicates that irradiated Zircaloy cannot readily accommodate stresses or strains of this kind, particularly when the Zircaloy has been exposed to certain embrittling (stress-corrosive) fission product species such as iodine or cadmium.

Fuel failures due to PCI were first recognized in late 1971 in a number of early reload BWR fuel assemblies that had operated for several cycles; a 1972 GE report (Ref. 27) subsequently identified PCI as a mechanism that could affect BWR fuel lifetime. Based on results of developmental investigations and feedback from production fuel experience, specific operating restrictions, known as Preconditioning Interim Operating Management Recommendations (PCIOMRs), were issued by GE to the BWR operators (Ref. 28).

PCIOMRs have generally been effective in reducing PCI failures that result from operational power changes, but they would not prevent PCI failures during unexpected transients and accidents. Several related criteria consisting of 1%-strain and centerline-melt limits are met for Zimmer, but these criteria are not effective in preventing corrosion-assisted PCI failues or highly localized-strain PCI failures. Hence, from a regulatory standpoint, there has been a deficiency in addressing the potential for PCI failure during power-increasing transients and accidents, and new techniques are being developed to address this deficiency (Ref. 29).

In conclusion, (a) the applicant will impose operating restrictions to reduce the potential for PCI, (b) the applicant has met several related criteria that preclude PCI failures under some conditions, and (c) the NRC is studying the need for new licensing requirements in this area. There are presently no other PCI licensing requirements that must be met for Zimmer.

Water Rod End-Plug Wear

Damaged fuel assembly components are sometimes not detected by monitoring equipment during plant operation. Detection, or confirmation of suspected damage is, however, often possible through visual inspections during refueling outages. One such fuel damage phenomenon that was first observed in the fall of 1979 involved water rod end-plug wear (Ref. 30). The wear occurred on the shanks of the water rod end plugs in 8x8R ("Retrofit") assemblies. The cause of the wear has been attributed to flow excitation of the water rods by coolant cross-flow within the lower tie plate flow volume.

The chief concerns associated with water rod-end plug wear are (a) the potential for loss of positive spacer positioning (due to rotation that could result from significant wear of the lower end plug), and (b) loose parts. A loose-parts analysis performed by GE was submitted in June 1980 (Ref. 31) and concluded that generation of a loose part, although not expected, would not result in unacceptable consequences. That is, "worst case" offsite doses would remain well below 10 CFR 100 guideline values, and there was no potential for control rod interference, unacceptable fuel assembly flow reduction, or chemical action on reactor internals.

The potential for loss of positive spacer positioning due to water rod end-plug wear was assessed with the aid of inspection results from two operating plants. Those results (Ref. 32) showed that wear of the spacer-positioning water rod will remain below the valve (~ 30 mils) at which rotation could theoretically occur. In addition, further analysis indicated that even if sufficient wear were to occur, there were insufficient rotational forces to cause spacer rotation. Therefore, based on the above surveillance and analytical results, we agree with GE that no further action is required on water rod end-plug wear at this time.

Water-Side Corrosion

As an adjunct to visual inspections, "sipping" is often performed with BWR fuel assemblies, particularly if offgas activity trends (at the steam jet air ejector) indicate that there are a significant number of "leakers" (sipping is basically the sampling of activity around a bundle and is a highly effective means for identifying leaking bundles). In the spring of 1979, several leaking BWR bundles were identified by sipping (Ref. 33) in a case which involved cladding failures caused by external "water-side" corrosion. Those failures have been associated with a variably high copper concentration in the core coolant water and a minor anomaly in the Zircaloy cladding metallurgy (Refs. 34, 35), although both the water chemistry and the cladding metallurgy were within allowable specifications. The source of the copper contamination in the affected plant was judged to be the copper-bearing main condenser tubes (Ref. 35c).

While Zimmer does have copper-bearing main condenser tubes, the design of the Zimmer condensate and feedwater system provides that all water that enters the reactor feedwater system is processed by the condensate polishing demineralizers (after passing through the high copper content condenser tubing, but before entrance into the feedwater cycle) (Ref. 36). The condensate polishing demineralizers are of the deep bed design and provide a maximum effluent copper content of two ppb, which is in conformance with the General Electric recommendation for this service. With a maximum of 2 ppb in the feedwater, the reactor water cleanup system should be able to maintain the cooper concentration in the reactor coolant inventory below the level at which copper deposition on the fuel rods will occur. Hence, in plants such as Zimmer, which has deep-bed demineralizers that can remove copper contamination resulting from the copper-bearing main condenser tubing, waterside corrosion failures of this kind would not be expected. Therefore, inasmuch as (a) a failure episode of this type has occurred only once in GE fuel operating history, (b) such failures are detectable, thus permitting remedial action, and (c) adequate preventive design measures appear to have been taken, we conclude that this issue has been satisfactorily resolved for Zimmer.

Hydriding

In addition to external water-side corrosion, internal "fuel-side" cladding corrosion has been a concern in BWR fuel and burnable poison rods. The principal internal corrosion problem has in the past been caused by hydriding (Ref. 37). The hydriding defects normally appear as localized hydride nodules or blisters, which may crack or form craters in the cladding. The primary source of hydrogen inside the rods has been identified as contamination by moisture in the UO_2 during manufacture. Identification of hydrogeneous impurities internal to the fuel and poison rods as the cause of hydriding failure led to the development of process controls to limit the introduction of such impurities and to remove any impurities that might inadvertently be introduced. In addition, GE uses a hydrogen-getter material in the upper plenum of all fuel rods. Thus, for the Zimmer fuel, a combination of design features and manufacturing controls will be used to ensure that the moisture content in a loaded fuel rod is below the threshold that would cause hydriding failure.

Rod Bowing

Irradiation-induced bowing in fuel rods and assemblies is a phenomenon which is not, in itself, a failure mechanism, and which is not limited to specific set values (i.e., acceptance criteria in the Standard Review Plan). It must be addressed, however, in the design analysis so as to establish operational tolerances. GE has asserted (Ref. 38) that BWR fuel operating experience, testing, and analysis indicate that there is no significant problem with rod bowing even at small rod-to-rod and rod-to-channel clearances. Specifically, GE noted that (a) no gross bowing has been observed (excluding the rod bowing-related failures in an early design); (b) a very low frequency of minor bowing has been observed; (c) mechanical analysis indicates deflections within design bases; and (d) thermal-hydraulic testing has shown that small rod-to-rod and rod-to-channel clearances pose no significant problem. Based on those reported observations and the recent submittal and ongoing review of a GE generic topical report (Ref. 39) that is expected to (a) update the GE rod bowing experience, (b) verify the accuracy with which GE measures rod bowing, and (c) document the overall GE rod bowing safety analyses, we conclude that there is currently no reason to anticipate a problem with fuel rod or assembly bowing during operation of Zimmer Unit 1. Should any future action be required as a result of our review of the submitted GE topical report on rod bowing, we will inform all BWR licensees of our requirements.

Channel Box Wear

In addition to the fuel rods, there are other fuel system components whose functionality must be assured as an objective of the review of fuel system design. One such component in BWR cores is the fuel assembly channel box. The fuel "channel" that encloses the fuel bundle performs three functions: (1) the channel provides a barrier to separate two parallel flow paths (one to cool the fuel bundle and the other to cool the bypass region between channels); (2) the channel guides the control rod and provides a bearing surface for it; and (3) the channel provides rigidity for the fuel bundle. Thus, the potential for cracks or holes in a "channel" or channel "box" is of concern since it would allow part of the cooling water that normally flows through a fuel bundle to flow out of the cracks or holes and bypass the fuel rods. Such a change in flow pattern would reduce the safety margin for fuel thermal performance and would lead to fuel overheating and damage in the event of some anticipated operating transients or postulated accidents. Significant channel box cracking and wear could also adversely affect mechanical strength.

In the mid 1970s, channel box wear and cracking was observed, first in a foreign plant and later in a few domestic BWRs. The wear was located adjacent to incore neutron monitor and startup source locations. It was postulated

(Ref. 40), and later confirmed by out-of-reactor testing, that the wear was caused by vibration of the incore tubes due primarily to a high-velocity jet of water flowing through the bypass flow holes in the lower core plate. To eliminate significant vibration of instrument and source tubes and the resultant wear on channel loop corners, Zimmer Unit 1 will incorporate modifications similar to those described (Ref. 41) for BWRs currently in operation. Those modifications involve the elimination of the bypass holes in the lower core plate and addition of two holes in the lower tie plate of each assembly to provide an alternate flow path. This design modification has been determined to have negligible adverse effects on the mechanical, thermal, and nuclear performance of the channel boxes, as is discussed in our generic safety evaluation on this subject (Ref. 42). Because channel box wear has been observed (Ref. 43) to have been significantly reduced in operating BWRs following the design modification, we conclude that there is reasonable assurance that channel box wear and cracking will not be a problem in Zimmer Unit 1.

Channel Box Deflection

Another potential life-limiting channel box phenomenon involves the deflections caused by long-term creep deformations. While fuel channel design and deformation under operating conditions are discussed in the Zimmer FSAR, that discussion is out of date. In September 1976, General Electric issued a generic report, NEDO-21254, on channel box mechanical design and deflection (Ref. 44) and GE responded to NRC questions in two supplements issued in 1977. The GE report documents (a) the fuel channel description, (b) its design bases and design analyses, and (c) the creep deflection phenomenon.

The design bases and analyses portions of that report document the channel box development and are, therefore, largely of historical interest, but the creep deflection phenomenon is relevant to Zimmer plant operation and results in the bulging out of the channel box in a way that reduces the size of the gap provided for control blade insertion. This dimensional change, if it were large, could interfere with control blade insertion, so channels are discharged before this deflection becomes excessive.

The GE report, NEDO-21354, describes a channel lifetime prediction method and a backup recommendation for periodic channel measurements, which consist of settling friction tests. Although we believe the applicant intends to perform these recommended tests, we can find no reference on the docket to NEDO-21354, no commitment to perform these tests, and no specification of the test interval (which was not provided in NEDO-21354). When this information is provided by the applicant, the channel box design and deflection issue will be resolved.

We will report on the resolution of this issue in a supplement to this safety evaluation.

Control Blade Stress Corrosion Cracking

The stress corrosion cracking of control blade tubing is another example of a failure phenomenon that has been observed (Ref. 45) in a BWR fuel system component that is not an integral part of the fuel assembly itself. In this case, hot cell examinations of both foreign and domestic control blades revealed cracks in some of the stainless steel tubing and a loss of boron (B_4C) from some of the tubes. The safey significance of boron loss is its impact on shutdown capability and scram reactivity. Although shutdown capability is demonstrated by shutdown margin tests after refueling, the calculated control blade worths used in the safety analysis are based on the assumption that no boron loss has occurred.

Based on the hot cell observation that there is no boron loss until 50% local B-10 depletion (burnup) is attained (Ref. 45), GE assessed the potential effect of the boron loss on shutdown capability, CPR reduction, and the consequences of control rod drop accidents and concluded that (a) control rod drop accidents are not sufficiently sensitive to reductions in scram reactivity to be affected by small boron losses before the end of the blade's design life; (b) there is a negligible effect on transient CPR reduction and MCPR limits for small boron losses; but (c) if any control blades have experienced more than 10% reduction in projected worth, the shutdown margin should be demonstrated (by testing) to be adequate.

The bases for GE's conclusions, including the hot cell examinations and calculational assumptions, were reviewed, and it was decided that the relationship between boron loss and B-10 depletion was sufficiently well understood to justify BWR operation on an interim basis provided that certain actions were taken by the licensees. Those actions, which include further analyses, shutdown margin tests, and destructive examinations, are described in detail in IE Bulletin No. 79-26, Revision 1 (Ref. 47), and written responses are required of all operating BWR plants including Zimmer (once operation begins). Since these actions will be taken at Zimmer, assurance will be maintained that control blade reactivity will not be significantly degraded by control blade stress corrosion cracking.

4.2.4 <u>Testing, Inspection, and Surveillance Plans</u>

As noted in Section 4.2 of the Standard Review Plan (Ref. 1), the fuel testing, inspection, and surveillance plans are reviewed for each plant. For testing and surveillance, the level of detail required in the SAR depends on (1) whether the fuel design is standard or new and (2) whether or not the overall testing and inspection plans are essentially the same as for previously approved plants. With regard to postirradiation examination (PIE) or surveillance, the requirements vary, depending not only on whether the fuel design is new or old, but also on the possible detection of unusual behavior or gross failure during operation.

The Zimmer FSAR contains a description of (1) the type of quality control inspections performed during manufacturing and (2) (in a more general sense) the plan for inspection and testing of irradiated rods. In the latter areas, GE relies on fission product monitoring, fuel sipping, and other onsite inspections as well as detailed postirradiation examinations in hot cells. Fuel performance results on highly precharacterized lead test assemblies (LTAs) are provided in several reports listed in NUREG-0633 (Ref. 48) for years prior to 1979. More recent results are provided in Reference 49. The lead test assemblies are utilized as one means of providing some confirmation of design adequacy or early warning of negative features of the design. Details on recent LTA programs are provided in a GE report (Ref. 50).

4.2.5 Fuel System Review Conclusions

Although most of the objectives of the fuel system safety review have been met, there are four confirmatory issues to be resolved prior to completing our review:

1. Supplemental ECCS calculations with NUREG-0630 models.
2. Periodic channel box deflection tests.
3. Combined seismic and LOCA loads analysis.
4. Fission gas release analysis at high burnup.

When these issues are resolved, we will be able to conclude that the fuel system of Zimmer Unit 1 has been designed such that (a) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (b) fuel damage during postulated accidents will not be so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures will not be underestimated for postulated accidents, and (d) core coolability will always be maintained, even after severe postulated accidents. The applicant will have provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. The applicant has also provided for testing and inspection of new fuel to ensure that it is within design tolerance. When the noted confirmatory issues on the fuel system design have been resolved, we will be able to conclude that the applicant has met all the requirements of the applicable regulations, current regulatory positions, and good engineering practice. We will report on the resolution of the confirmatory issues in a supplement to this report.

All applicable requirements related to the reactor fuel are described in Section 4.2, "Fuel System Design," of the Standard Review Plan (Ref. 1). The applicable regulations and regulatory guides are: 10 CFR 50 Appendix A (GDC-10) (Ref. 2); 10 CFR 50.46 (Ref. 50); 10 CFR 50 Appendix K (Ref. 52); Regulatory Guide 1.3 (Ref. 53); Regulatory Guide 1.4 (Ref. 54); Regulatory Guide 1.25 (Ref. 55); Regulatory Guide 1.77 (Ref. 56); and Regulatory Guide 1.126 (Ref. 57). Some of these requirements are satisfied in Chapter 15 of the Final Safety Analysis Report rather than in Section 4.2.

4.3 Nuclear Design

4.3.4 Summary of Evaluation

Point Kinetics Model

In NUREG-0528, we noted that calculations of void coefficient may not be conservative under certain transient conditions. A potential nonconservatism in the use of the point kinetics transient analysis model for pressurization transients was identified in the SER for the Wm. H. Zimmer Nuclear Power Station. Subsequently a series of tests were performed in the Peach Bottom Unit 2 reactor in support of the qualification effort for a transient core model with one-dimensional kinetics. These tests indeed showed that for some overpressurization transients the point kinetics model gave nonconservative results. Accordingly, we require that pressurization transients be calculated by the one-dimensional ODYN code. The use of the point kinetics code for other transients not involving rapid pressure changes is still acceptable.

The Peach Bottom tests are described in two EPRI reports, EPRI NP-563, "Core Design and Operating Data for Cycles 1 and 2 of Peach Bottom 2," and EPRI NP-564, "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2." The one-dimensional transient code ODYN is described in NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volumes 1, 2, and 3. NEDO-24154 also presents the results of the comparison with the Peach Bottom tests. The staff evaluation of the ODYN code is presented in a letter dated February 4, 1981 from R. Tedesco, NRC, to G. G. Sherwood, General Electric. This letter also outlines the acceptable procedures for implementing the code. The applicant recalculated the selected overpressurization event with the ODYN code (see subsection 4.4.1 below). This action closes out the open item regarding the point kinetics model.

4.4 Thermal and Hydraulic Design

In NUREG-0528 dated January 1979, the staff identified three items as requiring additional information from the applicant. The items identified are (1) Loose Parts Monitoring System (LPMS), (2) flow control valve interlock, and (3) thermal-hydraulic limit determination using ODYN methods. The purpose of this Supplemental Safety Evaluation Report (SSER) is to document the results of our evaluation of the LPMS description, the installation of the flow control valve interlock and the ODYN reanalysis, which are provided by the applicant in Amendments 68 and 69 to the Final Safety Analysis Report.

4.4.1 Evaluation

Loose Parts Monitoring System (LPMS)

Zimmer-1 will have a LPMS installed and operational before commercial operation. This system consists of one indicator assembly composed of ten loose-part channels and two vibration channels, all with latching alarm lights, channel selector, two audio monitors with separate speakers, and electronic test and reset capability. According to the applicant, this system meets the requirements of Regulatory Guide 1.133. Therefore, we conclude that the LPMS for Zimmer-1 is acceptable.

Flow Control Valve Minimum Position Interlock

In Appendix H of Amendment 68 of the FSAR, the applicant indicated that the flow control valve (FCV) was installed to prevent system startup or transfer from 25-100 percent speed unless the valve is in minimum position. According to the applicant, this is to prevent scram due to a rapid flow increase resulting from an operator failure to close the FCV prior to the start of speed transfer. We find that this is acceptable.

ODYN Reanalysis

The Minimum Critical Power Ratio (MCPR) limit originally proposed was based upon calculations using the REDY model described in NEDO-10802. During our review of the General Electric Company analytical methods described in NEDO-10802, three turbine trip tests were performed at the Peach Bottom Unit 2 boiling water reactor. The results from the tests revealed that in certain cases the results predicted by REDY are nonconservative. We reviewed this matter on a generic basis with the General Electric Company and approved a new calculation basis using the General Electric Company's new computer code ODYN. We required, as identified in the SER, the applicant to reanalyze the following transients for the MCPR determination: (1) feedwater controller failure - minimum demand, (2) generator load rejection, and (3) turbine trip.

We reviewed the applicant's reanalysis, included in Amendment 119, which provide the analytical results of limiting transients using ODYN methods. Results indicate that the operating limit MCPR, as a function of the scram time, is determined by using the maximum calculated transient MCPR. Based on our review, we find that (1) the approved ODYN methods were used and (2) the results of the analyses do not violate the safety limit MCPR of 1.06. Therefore, we conclude that the ODYN analysis is acceptable.

4.4.2 Conclusion

The thermal-hydraulic design of the core for Zimmer-1 was reviewed. The scope of the review included the design criteria, implementation of design criteria as represented by the final core design, and the steady-state analysis of the core thermal-hydraulic performance. The applicant's thermal-hydraulic analyses were performed using analytical methods and correlations that have been previously reviewed and found acceptable. However, the operating license should be restricted to the following conditions:

(1) part loop operation is not permitted unless supporting analyses are provided and approved for the second cycle of operation.

(2) operation beyond Cycle 1 is not permitted until a new stability analysis is provided and approved for the second cycle of operation.

We conclude that, with the restrictions noted above, the thermal-hydraulic design of the core conforms to the Commission's regulations and to applicable Regulatory Guides and staff technical positions as set forth in the Standard Review Plan Section 4.4 and is, therefore, acceptable.

4.5 Reactor Materials

4.5.1 Control Rod System Structural Materials

As discussed in NUREG-0528, during routine maintenance inspections of a General Electric Company-designed boiling water reactor in June 1975, dye penetrant inspections of control rod drive components revealed fine cracks in some of the control rod tubes. Subsequent inspections of other drives that had been in operation disclosed similar cracks. Conventional metallography and scanning electron microscopy identified the cracking as inter-granular in nature. The cracks were generally circumferential and appeared mainly where the wall thicknesses change in the area between the ports. The cracks had developed from the outside of the tube, but none of the cracks were through the wall. They were generally shallow, less than half the wall thickness. Many of the cracks were "tight" and filled with oxide.

Operating experience obtained from 270 boiling water reactor years and results of the 779 control rod tubes (out of about 4000 drives in service) inspected by the dye penetrant technique at 22 sites disclosed that partial cracking had occurred in 78 tubes at 11 of the sites.

We established a Category B Technical Activity (No. B-48) to address and resolve the issue of boiling water reactor control rod drive mechanical failures. This activity has been completed.

The control rod tube cracking that has occurred to date is generally shallow, intermittent, and very "tight," and has not impaired the control rod drive's ability to meet its functional requirements. The General Electric Company has proposed a new design which is acceptable to us. However, the applicant plans to utilize the old control rod drive design for initial operation. As a precautionary measure, we will require and the applicant has committed to augmented inservice inspection to assure early detection of cracking. We find the applicant's actions regarding this matter acceptable for licensing.

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 and further clarified these requirements in a letter dated March 30, 1981. This matter is the subject of IE Bulletins 80-14 and 80-17. The staff will review conformance of the Zimmer scram discharge volume design with its generic study presented in the December 1, 1980 report and provide its conclusions in a future supplement to this report.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

Reactor Vessel Supports

In NUREG-0528 we noted, "The applicant was requested to provide stresses and corresponding margins of safety for critical reactor vessel support components due to transient pressure loads resulting from worst case blowdown (see subsection 3.9.1 of this report). The applicant provided the requested information in Amendment 76 to the Final Safety Analysis Report, August 28, 1978."

We have reviewed this information and find that the applicant has provided the results of his evaluation of the reactor pressure vessel support loads resulting from the simultaneous occurrence of asymmetric subcompartment loss-of-coolant pressure and safe shutdown earthquake loads. The results verify that the reactor pressure vessel support loads are less than design loads for all structural support elements of the reactor pressure vessel. We find the applicant's analysis acceptable.

Feedwater and Control Rod Return Line Nozzle Cracking

In NUREG-0528 we noted, "We conveyed our position to the applicant describing an acceptable procedure for assuring early detection of possible occurrence of cracks in the feedwater nozzles, control rod drive return line nozzles, and vessel blend radii. The applicant will provide a satisfactory response to our position regarding the augmented inservice inspection program. We initiated discussions with the applicant pertaining to this matter. Resolution of this matter will be provided in a supplement to this report. (See subsection 5.2.4 of this report.)"

On November 13, 1980, the staff sent NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," to the applicant for implementation. The applicant will be expected to implement NUREG-0619 items requiring completion prior to the issuance of an operating license. Other items will remain as a condition of the operating license. The applicant is expected to implement the recommendations of NUREG-0619 to the extent practical prior to the issuance of an operating license.

5.2.2 Overpressurization Protection

In NUREG-0528 we noted, "Trip of the recirculation pumps at high vessel pressure is used to provide partial mitigation of the consequences of anticipated transients without scram. It was found that the effects of this trip were not included in the analyses. We require that overpressurization calculations, including the effects of this trip, be submitted for evaluation. The results will be discussed in a supplement to this report.

"The safety relief valve manufacturer tests the valves hydrostatically, for valve response, for set pressure, and for seat leakage prior to shipment to certify that design and performance requirements have been met. Specified manual and automatic actuation is verified during the preoperational test program. This complies with the preoperational testing of Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants." In addition, one-half of the valves will be tested to check set pressure at each refueling outage. We requested information concerning qualification tests and operating experience with the safety relief valves with respect to the safety mode of activation and performance of the automatic depressurization function. The results of our review of the response to this information request will be discussed in a supplement to this report."

We have completed our review of these matters and have concluded the following.

We required the applicant to recalculate the overpressurization transient using the reactor pump trip to assure that the most conservative event has been considered. The applicant's analysis showed that the overpressurization transient resulting from main steam isolation valve closure does not violate the reactor coolant system pressure criteria when the high neutron flux reactor scram is used with recirculation pump trip (for this event, high reactor pressure is used as a signal for reactor pump trip, but would occur following the high neutron flux signal).

The overpressurization analysis was performed using the computer-simulated model described in General Electric Topical Report NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the GE BWR." Comparison of the REDY Code (NEDO-10802) with turbine trip tests at Peach Bottom showed the REDY code to be nonconservative for overpressurization events. We have reviewed this matter on a generic basis with the General Electric Company and have evaluated a new calculational basis using the General Electric Company's new computer code ODYN (Ref: Letter from D. G. Eisenhut to holders of CP and OL for BWRs dated January 29, 1981).

At our request, the applicant has performed overpressurization analysis using ODYN. The calculated peak pressure for high neutron flux scram using ODYN was 1270 psig (i.e., 30 psig less than the pressure calculated by REDY). Therefore the peak calculated pressure remains below the acceptance criterion of 110% × design pressure and is acceptable.

With regard to information concerning the qualification tests and operating experience of safety/relief valves, the applicant has provided this information in Revision 51, dated January 1979, to the FSAR. The applicant states that there have been no malfunctions reported for safety/relief valves used in the Chin Shan I BWR, an operating plant which has safety valves of the same type as Zimmer (Crosby). Life Cycle and environmental qualification tests for Crosby valves were conducted. The acceptability of these tests will be discussed in Sections 3.9.1, 3.10, and 3.11 to the SSER. In addition, the applicant is participating in the BWR Owners Group Program for performance testing of safety/relief valves (Item II.D.1 of NUREG-0737).

It is noted that the lowest setpoint of safety/relief valves is changed from the 1165 psig indicated in NUREG-0528 to 1150 psig (in the safety mode). It is also noted that the General Electric Company has agreed to work with the staff and their utility customers to maintain a surveillance program once new safety-relief valves become operational (NUREG-0152). Information to be reported will include all abnormalities ranging from minor wear observed during normal inspection to complete failures, including failure to open or close and inadvertent operation. The applicant has committed to participate in this program.

We have reviewed the system designed to prevent overpressurization of the reactor coolant system. We conclude that this system conforms to the requirements of Criterion 15 of the General Design Criteria and the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and is acceptable.

5.2.3 Reactor Coolant Pressure Boundary Materials

Stainless Steel Pipe Cracking

Since mid-1960, leaks and cracks have been observed in the heat-affected zones (HAZs) of welds that join austenitic stainless steel piping and associated components in BWRs. The systems and components where cracking has occurred includes recirculation bypass lines, control rod drive (CRD) hydraulic lines, isolation condenser lines, recirculation inlet lines where crevices are formed by the welded thermal sleeve attachment, shutdown heat exchanger lines, and core spray lines. The cracking mechanism is attributed to intergranular stress-corrosion cracking (IGSCC) that resulted from a combination of high residual stress, sensitized material, and high oxygen content in the coolant.

In NUREG-0528 we stated that we had requested the applicant to provide us with information regarding his implementation of our position stated in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping." We had reviewed this information and concluded that the applicant's implementation of the position in NUREG-0313 was acceptable. However, Revision 1 to NUREG-0313 was issued, and on January 18, 1980, by letter, the applicant was requested to respond to the revision. On February 26, 1981, by letter, the applicant was informed of the staff's implementation position.

NUREG-0313 Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," documents the NRC staff's revised guidelines for reducing the susceptibility of BWR piping to IGSCC and provides an increased level of reactor coolant pressure boundary and engineering safety features systems integrity.

Applicants are requested to review all ASME Code Class 1, 2, and 3 pressure boundary piping, safe ends and fitting material, including weld metal, at their facility to determine if the material selection, processing guidelines, or inspection requirements set forth in the report are satisfied. Until the applicant submits information demonstrating compliance with NUREG-0313 Rev. 1, our evaluation of the potential for IGSCC in BWR coolant pressure boundary piping will remain an open item. Full implementation of the guidelines will be made a condition of the operating license.

Fabrication and Processing of Ferritic Materials

By letter, dated May 20, 1980, the applicant was advised that applicants will be required to demonstrate the adequacy of the applicable support structures of their facilities from a fracture toughness standpoint. In the case of Zimmer the applicable support structures are the reactor vessel and reactor coolant recirculation pump supports. The letter provided additional guidance on "Potential For Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports" - NUREG-0577. The applicant has not responded to date, but completion of the staff's review is not required prior to a licensing date of October 1981. By letter of October 6, 1980, the applicant was informed of the staff's implementation position which is as follows.

"As you are aware, NUREG-05[illegible] (Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports) was issued in its "For Comment" form in November 1979. Additional guidance regarding implementation was issued by letters dated May 19, 1980 (licensees) and May 20, 1980 (applicants). A significant difference between NUREG-0577 and the criteria contained in the May 19 and 20 letters was the removal, in the letters, of the option of using linear elastic fracture mechanics analyses to demonstrate adequate structural integrity. The largest single response by applicants and licensees to the May 19 and 20 letters indicated a desire to retain linear elastic fracture mechanics analyses as an alternative means to demonstrate adequate structural integrity. Because of the response received due to this change, the NRC staff convened a meeting on August 27, 1980 to resolve the differences in the proposed programs. You should especially note these items of importance: (1) No action is required on your part at this time regarding lamellar tearing; (2) the subject of irradiation effects on reactor vessel support materials is under review by the NRC staff and is not part of the A-12 effort; and (3) No implementation action on your part will be required before the December 1980 meeting. Exceptions to this policy may be necessary on a case-by-case basis if significant materials problems are found to exist. Of more importance, however, is our request that you be prepared to commit to the alternative program if approved by the NRC staff at the meeting to be held in December 1980. Failure to commit to the alternative program will result in NRC imposition of the guidance contained in May 19, 1980 and May 20, 1980 letters, as modified by applicable comments under review by the NRC staff."

The recommendation for extension of the implementation period is still under staff review. Resolution of this issue will be made a condition of the operating license.

Fabrication and Processing of Austenitic Stainless Steel

By letter, dated June 6, 1980, the applicant was requested to provide additional information regarding observed cracking of BWR jet pump holdown beams. This issue is the subject of IE Bulletin 80-07. The applicant responded to the June 6, 1980 request by letter, dated October 31, 1980.

The applicant provided the following information.

Evaluations performed by General Electric (GE) have determined that beam failures have resulted from intergranular stress corrosion cracking. A comparison of the failed BWR 3 beam with the BWR 4-6 beam, as used on Zimmer, indicates that the Zimmer beam operates at a peak stress 14% lower than the BWR 3 beam at the present preload. Since time to failure is dependent on applied stress, the BWR 4-6 beams, as presently designed and installed, are predicted to have a longer life.

A reduction of the 30 kip preload currently specified for BWR 4-6 beams to 25 kip will yield a significant factor of improvement in predicted time to crack initiation. Using relationships developed from field experience and laboratory stress corrosion tests, minimum time to crack initiation of the Zimmer jet pump beam is estimated to increase by at least a factor of four with respect to the BWR 3 jet pump beam. Additional testing to be conducted through 1981 should make it pssible to more accurately predict the expected life of beams with preload reduction. The operational acceptability of this reduced preload has been demonstrated by tests in the GE high flow test facility. Based on current test data, the preload reduction is expected to increase the beam operating time to crack initiation, at a 2.5% probability level, to a range of 19 to 40 years.

Based on information available to date, the preload reduction as described above is expected to be a long-term solution for BWRs 4-6 and would be adopted by Zimmer.

If a long-term solution is not agreed upon, periodic inspections of the Zimmer holddown beams would be determined on the basis of results obtained from operating plants of the same design with surveillance programs currently in place. Crack propagation rates are slow enough to be readily detected during scheduled outages without fear of undetected cracks developing into breaks between outages. Surveillance via ultrasonic inspection can detect cracks in jet pump beams and would likely be employed on Zimmer.

The staff is reviewing the above response but finds the applicant's approach acceptable since the applicant has agreed to carry out an inservice inspection program satisfactory to the staff. If a deteriation of the holddown beams is detected by the inspection procedures, the applicant will replace the beams with improved design.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection & Testing

The staff has reviewed the Preservice Inspection Program for the Wm. H. Zimmer Nuclear Power Station Unit No. 1 for compliance with paragraph 50.55a(g) of 10 CFR Part 50.

Subsubarticle IWB-1220(b)1 of Section XI of the ASME Code provides for the exemption from preservice and inservice examination of components if "under the postulated condition of loss of coolant from the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system only." The makeup is defined in the Code as a system that has "the capacity to maintain reactor coolant inventory under respective conditions of startup, hot standby operation or cooldown, using onsite power."

The Cincinnati Gas and Electric Company elected to seek exemption under Subsubarticle IWB-1220(b)1 for piping from preservice volumetric examination according to the analysis presented in General Electric Company document 22A2756, Section 3.2.2, Revision 2. According to the analysis for the Zimmer Unit 1 facility, the exempted pipe was 2.12 inches for water and 4.24 inches for steam inside diameter, respectively.

The Reactor Systems Branch reviewed the reactor coolant makeup capability under loss-of-coolant accident conditions for the Zimmer Unit 1 facility and calculated the pipe sizes exempted under Subsubarticle IWB-1220(b)1, using the approved blowdown model referenced in Appendix K of 10 CFR Part 50. The condensate and control rod return flows were excluded in the calculation because they require offsite power for actuation. The limiting inside diameters were calculated to be 1.11 inches for water and 2.22 inches for steam, respectively.

The Cincinnati Gas and Electric Company indicated that there are ten additional pipe lines, containing approximately 170 welds, not excluded by Subsubarticle IWB-1220(b)1 by the Reactor Systems Branch calculations. The Cincinnati Gas and Electric Company agreed to examine the additional pipe in compliance with the requirements of Section XI, 1977 Edition, including Summer 1978 Addendum.

We have concluded from our evaluation that the Preservice Inspection Program for the Zimmer Unit 1 facility will be conducted in compliance with the requirements of Section XI, 1974 Edition and 1977 Edition, including Summer 1978 Addendum, of the ASME Boiler and Pressure Vessel Code to the extent practical. We have determined that certain Code requirements are impractical to perform and that relief is required and justified.

We have evaluated the alternate inspection procedures that will be performed in lieu of the specific Section XI requirements. Relief is justified because that alternate inspection procedures will provide an adequate level of quality and safety. Based on the results of our review, we have concluded that the Preservice Inspection Program for the Wm. H. Zimmer Nuclear Power Station, Unit No. 1, is in compliance with paragraph 50.55a(g) of 10 CFR Part 50.

Our safety evaluation is as follows.

The construction permit for the Wm. H. Zimmer Nuclear Power Station, Unit No. 1 was issued on October 27, 1972. Pursuant to paragraph 50.55a(g)2(ii) of 10 CFR Part 50, the preservice inspection requirement for Class 1 and 2 components should meet the examination rules set forth in Section XI, 1971 Edition, including Summer 1971 Addenda, of the ASME Boiler and Pressure Vessel Code.

Although this edition of Section XI did not include examination rules for the inspection of Class 2 components, the intent of the 1971 Edition of Section XI was to perform the preservice examination as closely representative as practical to the examination to be performed later during operation of the facility. In addition, the preservice examination might be performed in the fabrication shop if the method and technique of examination were the same as those expected to be used for the inservice examination.

The provision of paragraph 50.55a(g)3(v) of 10 CFR Part 50 permits the use of later editions and portions thereof of Section XI of the ASME Code, subject to certain limitations and modifications. The Cincinnati Gas and Electric Company elected to select the examinations, procedures, and acceptance criteria of Section XI, 1974 Edition of the ASME Code for the Preservice Inspection Program

Plan for the Zimmer Unit No. 1 facility. All Class 1, 2, and 3 components and their supports are scheduled to be examined and to comply to the extent practical with the requirements of this edition of the Code.

Augmented inservice inspection of components required by the NRC have been conducted at the Zimmer Nuclear Power Station No. 1. However, the details of the augmented examinations were not incorporated in Preservice Inspection Program Plan but will be documented in the revised program.

Features were incorporated in the design of components to provide access to implement the preservice and inservice examination requirements of Section XI. In addition, welds that might be inaccessible after installation were examined prior to component setting in order to provide essentially complete documentation of the examination requirements.

The Cincinnati Gas and Electric Company determined that conformance of certain requirements of the Section XI, 1974 Edition of the ASME Boiler and Pressure Vessel Code were impractical to conduct at the Wm. H. Zimmer Nuclear Power Station Unit No. 1 facility. The requirements held to be impractical were described in Appendix A, Exemptions and Exceptions, Document 80A1181, Preservice Inspection Program Plan, prepared by the Nuclear Energy Services, Inc. Pursuant to paragraph 50.55a(g)(6)(i) of 10 CFR Part 50, the NRC was requested to evaluate the Preservice Inspection Program Plan and grant relief from the Code examination requirements that were held to be impractical. The purpose of this supplement is to review the request and to evaluate the bases for granting relief from the Code requirements.

Request No. 1 - Integrally Welded Supports - Table 1WB-2600; Category B-K-1; Item Nos. B4.9, B5.4, B6.4

Code Requirement - Volumetric examination of 25% of the integrally welded supports is required during each inspection interval. This examination includes the welds to the pressure-retaining boundary and the base metal beneath the weld zone and along the attachment member for a distance of two support thickness.

Code Deviation Requested - Substitute surface for the volumetric examination requirement.

Basis for the Relief Request - Shear lugs and other permanent attachments were fabricated on the pressure-retaining components of the Zimmer Unit 1 facility to the requirements of Section III, 1971 Edition of the ASME Code. The welds are full penetration welds examined by surface examination procedure in compliance with paragraphs NB-4433 and NB-5266 of Section III, respectively. Due to the geometric configuration, all these welds are not amenable to volumetric examination as required by Section XI of the Code. The welds are attached to but do not penetrate the pressure boundary.

Each integrally welded support will be examined in compliance with the requirement of Section XI of the Code and evaluated on a case-by-case basis. The applicant has committed to perform a surface examination on those welds where it is impractical to perform a volumetric examination in compliance to Section XI of the ASME Code.

Evaluation - We concur that a best effort preservice inspection program plan to perform a volumetric examination of the welds of the integrally welded supports, augmented by a surface examination where the geometric configuration limits the practicality of the volumetric examination, will provide adequate assurance of structural integrity of the supports. Relief is granted from the requirement of Section XI, 1974 Edition of the ASME Code. Granting of this requested relief is consistant with the requirement of Section XI, 1977 Edition, including Summer 1978 Addenda, of the ASME Code and comply with the latter Code requirements.

Request No. 2 - Pressure-Retaining Bolting Exceeding 1-inch in Diameter - Table 1WC-2520; Category C-D; Item Nos C1.4, C2.4, C3.2, C4.2

Code Requirement - The 1974 Edition of Section XI requires that visual and either surface or volumetric examination be performed on pressure-retaining bolting exceeding one inch in diameter. Visual examination is required during each inspection interval on 100% of the bolts, studs, nuts, bushing and threads in the base material and the flange ligaments between threaded nut holes. Surface or volumetric examination is required on 10% of the bolting in each joint, but not less than two bolts per joint.

Code Deviation Requested - Substitute two-inch-diameter for the one-inch-diameter examination criterion.

Basis for the Relief Request - Relief is requested from the one-inch-diameter criterion and the substitution of the two-inch-diameter criterion of the Summer 1976 Addenda of Section XI of the ASME Code. The bases for the relief request are that the Summer 1976 Addenda is acceptable to the NRC staff and that examination of bolting less than two inches in diameter would impose a hardship without a commensurate gain in component integrity.

Evaluation - Pursuant to paragraph 50.55a(b)(2)(i) of 10 CFR Part 50, the provisions of the Summer 1976 Addenda of Section XI of the ASME Code are not acceptable to the NRC staff and can not be used to justify this request for relief. However, the provisions of Section XI, 1977 Edition, including Summer 1978 Addenda, are acceptable and may be used in support of this relief request.

The latter Code edition requires volumetric examination of 100% of the pressure-retaining bolting two inch in diameter and greater in accordance with Figure 1WC-2520-6. Visual and surface examination procedures are not required. In addition, the latter edition of the Code does not provide for the examination of Class 2 pressure-retaining bolting less than two inches in diameter.

We concur with the applicant that the one-inch-diameter criterion of the 1974 Edition of Section XI of the ASME Code would impose a hardship and burden without a commensurate increase in component integrity. In lieu of the Section XI, 1974 Edition, requirement, including the one-inch-diameter examination criterion, the preservice inspection of pressure-retaining bolting exceeding two inches in diameter should be examined to the requirement of Section XI, 1977 Edition, including Summer 1978 Addenda, of the ASME Code. An augmented visual examination (VT-2) is required by the staff on the exempted one-inch-diameter pressure-retaining bolting. The augmented visual examination (VT-2) should be conducted in compliance to paragraph 1WA-5240 of Section XI, 1977 Edition, including Summer 1978 Addenda, of the ASME Code.

Conclusion

We conclude from our review of the Preservice Inspection Program Plan for the Wm. H. Zimmer Nuclear Power Station Unit No. 1 that the preservice inspection will be conducted to comply to the extent practical with the examination requirements of Section XI, 1974 Edition, of the ASME Boiler and Pressure Vessel Code.

The Cincinnati Gas and Electric Company determined that certain requirements of Section XI of the ASME Code were impractical to conduct and requested relief from the Commission. The impractical examination requirements were described in Appendix A, Document No. 80A118, Nuclear Energy Services, Inc., March 31, 1978, and other addenda and submittals to the FSAR, including Amendment No. 66. We have reviewed and evaluated the request for relief from certain requirements of Section XI of the ASME Code for the Wm. H. Zimmer Nuclear Power Station Unit No. 1. We concur with the applicant that certain preservice examination requirements were impractical to conduct and relief was required.

We have reviewed the alternate methods of examination proposed by the applicant in lieu of the impractical requirements, and conclude that an adequate margin of safety will be provided. Pursuant to paragraph 50.55a(g)(6)(i) of 10 CFR Part 50, we have granted relief from the specific requirements identified to be impractical for the facility, giving due consideration to the burden placed upon the applicant if the specific code requirement was imposed, and which we have determined that by granting such relief will not endanger life, property or common defense and security of the public. Based on our review and evaluation, we concluded that the preservice inspection program for the Wm. H. Zimmer Nuclear Power Station No. 1 meets the requirements of paragraph 50.55a(g) of 10 CFR Part 50.

The conduct of the preservice inspection and hydrostatic testing of pressure-retaining components in compliance with the requirements of Section XI of the ASME Code will provide reasonable assurance that structural degradation or loss of leaktight-integrity which may occur inservice will be detected before the safety function of the component is compromised. Compliance with the preservice examination required by Section XI of the ASME Code in conformance to paragraph 50.55a(g) of 10 CFR Part 50 constitutes an acceptable basis for satisfying the requirements of General Design Criterion No. 32.

Inservice Testing of Pumps and Valves

The applicant has submitted a description of its proposed inservice testing program for pumps and valves. The program includes both baseline preservice testing and periodic inservice testing. It provides both for functional testing of components in the operating state and for visual inspection for leaks and other signs of degradation.

The date of the construction permits, October 27, 1972, places this facility under 10 CFR Part 50.55a(g)(2) which requires design and access to comply with the 1971 Edition of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code through the Winter 1972 Addenda. Since inservice testing requirements for pumps and valves were not included in the Code until the Summer 1973 Addenda of the 1971 Edition, the applicant has chosen to optionally meet the requirements of the 1977 Edition through the Summer 1978 Addenda to the extent practicable and has requested relief from the certain Code requirements.

We have not completed our detailed review of the applicant's submittal. However, based on our preliminary review, we find that it is impractical within the limitations of design, geometry, and accessibility for the applicant to meet certain of the American Society of Mechanical Engineers Code requirements. Imposition of those requirements would, in our view, result in hardships or unusual difficulties without a compensating increase in the level of quality or safety. The relief requested will not endanger life or property and is the public interest. Therefore, pursuant to 10 CFR Part 50.55a, the relief that the applicant has requested from the pump an valve, testing requirements of 10 CFR Part 50, Section 50.55(g)(2) and (g)(4)(i) is granted for that portion of the initial 120-month period during which we complete our review. Since the applicant's request for relief has been granted and the applicant will comply with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and/or the Technical Specifications, we find the Zimmer inservice testing program for pumps and valves acceptable.

One area of concern is the periodic leak testing of pressure isolation valves. There are several safety systems connected to the reactor coolant pressure boundary that have design pressures below the rated reactor coolant system pressure. In order to protect these systems from overpressure, two or more isolation valves are placed in series to form a pressure boundary interface between the high and low pressure systems. The leaktight integrity of these valves must be insured by periodic leak testing to prevent an intersystem LOCA and the possible overpressurization of the low pressure systems.

The applicant's response to Question 212.58 is not satisfactory. We will require that pressure isolation valves for the low pressure/high pressure core spray, residual heat removal, and reactor coolant isolation cooling systems be categorized as Category A or AC. Pressure isolation valves are required to be Category A or AC and to meet the appropriate valve leak rate test requirements of IWV-3420 of Section XI of the ASME Code. The allowable leakage rate shall not exceed 1.0 gallon perminute for each valve as stated in the Standard Technical Specifications. The applicant will be required to meet leak test requirements as specified in the Standard Technical Specifications NUREG-0123 Revision 3-3/4.4.3.

On receipt of the applicant's commitment to categorize and leak test pressure isolation valves as discussed above, we will conclude that the applicant's program provides reasonable assurance that the design pressure of low pressure systems will not be exceeded and that the probability of an intersystem LOCA has been reduced in accordance with the requirements of General Design Criterion 55.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

General Design Criterion 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed with sufficient margin to assure that when stressed under operating, maintenance, and testing conditions the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. General Design Criterion 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A, 10 CFR Part 50, requires, in part, that the reactor coolant pressure boundary be designed to permit an appropriate material surveillance program for the reactor pressure boundary.

The reactor vessel for William H. Zimmer Nuclear Power Station Unit 1 was ordered November 1969 and fabricated by Chicago Bridge & Iron to the 1968 ASME Code, Summer 1970 Addenda (except N-335). The Construction Permit for Zimmer Unit 1 was issued in October 1972. The Edition and Addenda of the ASME Code applicable to the design and fabrication of any reactor vessel is specified in Section 50.55a of 10 CFR Part 50. Based on the reactor vessel order date, and the Construction Permit date, this section of the Code of Federal Regulations requires that the Zimmer Unit 1 reactor vessel meet the requirements of at least the 1968 Edition of the ASME Code, including Addenda through Summer 1970. Therefore, the applicant did comply with the explicit requirements of Paragraph 50.55a(c)(2), 10 CFR Part 50. Pursuant to paragraph 50.55a(c)(2) of 10 CFR Part 50, we have evaluated the reactor vessel ferritic materials in accordance with the 1968 Edition of the ASME Code through 1970 Summer Addenda.

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Materials Surveillance Requirements," of 10 CFR Part 50, specify the fracture toughness requirements for the ferritic materials of the reactor coolant pressure boundary during normal operation, testing, maintenance, and anticipated transient conditions.

Compliance to Appendix G, 10 CFR Part 50.

We have evaluated the information in the applicant's FSAR to determine the degree of compliance with the fracture toughness requirements of Appendix G, 10 CFR Part 50. Our evaluation indicates that the applicant has met all requirements of Appendix G, 10 CFR Part 50, except for Paragraphs III.B.3, III.B.4, III.C.2, IV.A.1, IV.A.3, and IV.B.4, for which the applicant has supplied sufficient information to justify exemptions. Our evaluation of deviation from the explicit requirements of these paragraphs is contained in the following section.

Paragraph III.B.3 of Appendix G requires that the temperature instruments and Charpy test machines be calibrated in accordance with Paragraph NB-2360 of Section III of the ASME Code. Verification of this required calibration was impossible since the testing organization only retained the calibration report until the next calibration. However, General Electric has stated that the test instruments and machines were routinely calibrated on a periodic basis. Based on the standard practice of this period and on past experience with Charpy testing, we conclude that it is very unlikely that the test instruments and machines were not adequately calibrated and that an exemption to the requirement for maintaining the calibration report is justified.

Paragraph III.B.4 of Appendix G requires that the testing personnel shall be qualified by training and experience and should be able to perform the tests in accordance with written procedures. For Zimmer Unit 1 material testing, no written procedures were in existence. However, the individuals were qualified by on-the-job training and past experience. Because these tests are relatively routine in nature and are continually being performed in the laboratory that conducted these tests, it is unlikely that the tests were conducted improperly. Consequently, we conclude that an exemption for not performing the tests in accordance with written procedures is justified.

Paragraph III.C.2 of Appendix G, 10 CFR Part 50, requires, in part, that the base materials used to prepare test specimens for the reactor vessel beltline region shall be from excess base plate from the vessel beltline region. Paragraph III.C.2 of Appendix G was not complied with in that materials used to prepare weld test specimens for the reactor vessel were taken from simulated weldments prepared from excess production plate. However, the weld wire and flux materials used in the test specimens are the same as those used in the reactor vessel beltline. After weld completion, the sample weldments were subjected to a heat treatment to obtain metallurgical effects equivalent to those produced during fabrication of the reactor vessel. Based on our evaluation of this information, we conclude that although the same base material was not used to prepare the test samples, an exemption from the specific requirements of Paragraph III.C.2 of Appendix G is justified because the same heat treatment, weld wire, flux, and welding process used in the vessel welds were used in the test specimens. Since the weld toughness properties are determined primarily by heat treatment, weld wire, flux, and welding process, and not by differences in similar base materials, the use of weldment test specimens having the same weld wire, flux, and heat treatment as the vessel welds is sufficient to satisfy the requirements of Paragraph III.C.2 of Appendix G and provides acceptable justification for an exemption to the exact requirements of Paragraph III.C.2 of Appendix G.

Paragraph IV.A.1 of Appendix G requires that reactor coolant pressure boundary (RCP) materials be tested to the requirements of NB-2330 of the ASME Code. Paragraph NB-2330 of the ASME Code requires that a reference temperature, RT_{NDT}, be determined for each ferritic material of the RCPB and that this reference temperature be used as a basis for providing adequate margins of safety for reactor operation. The value of RT_{NDT} is defined in the ASME Code as the higher of either (a) the nil ductility temperature (NDT), as determined by the dropweight test, or (b) a temperature of 60°F less than the temperature at which 50 ft-lb energy and 35 mils lateral expansion is achieved, as determined by the CVN impact test. In addition, the CVN impact test for base metal is to be conducted using specimens oriented in the transverse direction.

The applicant has not complied with the requirements of Paragraph NB 2330 of the ASME Code for reactor vessel base metal because the CVN impact tests were conducted with longitudinally oriented specimens instead of transversely oriented specimens. The applicant has not complied with the requirements of Paragraph NB-2330 of the ASME Code for reactor vessel weldmetals because (a) the CVN impact tests were not performed over a sufficient temperature range to determined the temperature at which 50 ft-lbs and 35 miles lateral expansion would occur and b) drop weight test were not performed. For reactor vessel weld metals and base metals which has not been drop weight and CVN impact tested per NB 2330 of the ASME Code, the applicant utilized correlations in General Electric Report Y1006A006 to extrapolate the existing data to determine the RT_{NDT}.

We have reviewed the data from Zimmer Unit 1 vessel material, WRC Bulletin 217, Electric Power Research Institute Reports, EPRI NP-121 Vol. II, April 1976, and EPRI NP-933, December 1978, and other reactor vessels under construction by the Nuclear Steam Supply System (NSSS) vendor. Our review of these data indicates that the correlations used by the applicant to determine the effect of specimen orientation and the temperature at which 50 ft-lbs would be achieved, are conservative.

However, the assumption by the applicant of a -50°F NDT temperature for welds is not conservative. Data submitted by the applicant for simulated welds which were fabricated using the same process, wire type, flux type, post weld heat treatment and fabricated by the manufacturer of the original beltline welds indicates that a conservative NDT is -20°F. The maximum estimated end of life (EOL) RT_{NDT} for the beltline welds would be +50°F. A weld with an initial RT_{NDT} of -20°F and a EOL RT_{NDT} of 50°F is not the limiting material in the beltline region since beltline metal plate C-7158-1 had an initial RT_{NDT} of +12°F and an estimated EOL RT_{NDT} of +85°F.

The applicant has not complied with the explicit requirements for determining the RT_{NDT} for ferritic base metal and weld metal in the reactor coolant pressure boundary (RCPS) because the material was procured prior to publication of Appendix G, 10 CFR Part 50. However, based on the above data and analysis, an exemption to testing all RCPB materials to the requirements of Paragraph IV.A.1 of Appendix G, 10 CFR Part 50 as detailed in Paragraph NB-2330 of the ASME Code is justified.

Paragraph IV.A.3 of Appendix G requires, in part, that materials for valves meet the requirements of Paragraph NB-2332 of the ASME Code.

According to Zimmer Unit 1 FSAR, the main steam isolation valves (MSIV) were purchased to the 1970 Draft ASME Code for Pumps and Valves. The material for the MSIV bodies were not CVN impact tested. According to Paragraph 50.55a, 10 CFR Part 50, the valves on Zimmer Unit 1 should have been purchased to the Winter 1970 Addenda to the 1968 ASME Code due to the October 1972 construction permit date. The Winter 1970 Addenda to the 1968 ASME Code requires that the material from which the main steam isolation valves (MSIV) bodies are made SA-216 Grade WCB, exhibit an average of 20 ft-lb CVN test energy for three tests (one test result no lower than 15 ft-lb) at 60°F below the lowest service temperature (in this case 72°F).

The applicant has supplied CVN data for main steam isolation valves bodies installed in six other plants which had been fabricated to the same specification and heat treated to an equivalent microstructre as the Zimmer Unit 1 MSIV. The data indicates that a conservative estimation of the temperature at which 20 ft lbs would occur for these materials would be 60°F. Based upon a 60°F test temperature, the lowest service temperaure is required to be 120°F. The effect of 120°F lowest service temperature for the MSIV is discussed in section 5.3.2 of the SER.

An exemption to CVN impact testing of MSIV bodies is justified since the applicant has supplied data which indicates the material would have met Paragraph IV.A.3 requirements for CVN impact testing.

Paragraph IV.B of Appendix G requires that the reactor vessel beltline materials have a minimum upper shelf energy, as determined by Charpy V-notch impact tests on unirradiated specimens in accordance with Paragraph NB-2322.2(a) of the ASME Code, of 75 ft-lb, unless it can be demonstrated to the Commission by appropriate data and analyses that lower values of upper-shelf energy still provide adequate margin for deterioration from irradiation.

In accordance with 10 CFR 50.55a, the fracture toughness tests were conducted to an ASME Code Edition that preceded the effective data of Appendix G to 10 CFR Part 50. This Edition of the ASME Code did not require that the upper-shelf energy be established but only required that the tests be conducted at a single temperature equal to 60°F below the lowest service temperature. The test temperature determined in this manner typically was 10°F. However, all of the reactor vessel beltline plate materials were also tested at higher temperatures.

All the reactor vessel beltline materials tested met the minimum 75 ft-lb upper-shelf requirement except for the following:

1. Plates: C7185-1, C7185-2, C7151-1, C7158-1

2. Welds: (Seam/Heat Number/Linde Flux Lot Number)
   E1/3986/3876, E2/3986/3876, DE/04P046/D217A27A, DE/05P018/D211A27A

The applicant has provided CVN impact test data over a temperature range for base plates which had been procured to the same specification and heat treated to an equivalent microstructure as the Zimmer beltline plates. The applicant has also provided CVN impact test data over a temperature range for weld metal which had been fabricated using the same process, wire type, flux type, post weld heat treatment and fabricated by the manufacturer of the original beltline welds. These data show that although the CVN impact test results are below 75 ft-lb at +10°F, the upper shelf CVN values for both the plates and welds would be above the 75 ft-lb required by paragraph IV.B of 10 CFR Part 50. Thus an exemption to the explicit requirements of paragraph IV.B is justified.

Compliance to Appendix H, 10 CFR Part 50

The toughness properties of the reactor vessel beltline materials will be monitored throughout the service life of Wm. H. Zimmer Unit No. 1 by a materials surveillance program that must meet the requirements of Appendix H, 10 CFR Part 50. We have evaluated the applicant's information for degree of compliance to these requirements and have concluded that the applicant has met all requirements of Appendix H, 10 CFR Part 50, except for Paragraph II.B, for which the applicant has supplied sufficient information to justify an exemption. Our evaluation of the deviations from the explicit requirements of Paragraph II.B follows.

Paragraph II.B of Appendix H requires, in part, that the surveillance program for the ferritic materials in the reactor vessel beltline comply with ASTM E 185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessel." ASTM E 185-73 requires that the limiting reactor vessel beltline materials be included in the surveillance program that the CVN specimens be oriented in the transverse direction, and that each capsule contain at least 12 CVN impact specimens from each base metal, heat affected zone and weld metal sample. The samples from the heat affected zone are required to be removed from the limiting base metal plate.

According to our evaluation, plate C7158-1 is the most limiting base material and weld 1NMM/S-3986/3876 the most limiting weld metal. The Zimmer Unit 1 surveillance program contains materials from plate C7151-1 and weld 1NMM/KN203/0171. Because the Zimmer Unit 1 surveillance materials are not the most limiting base plates and welds, the applicant's materials surveillance program is not in full compliance with Appendix H, 10 CFR Part 50. To have an acceptable surveillance program for Zimmer Unit 1, the applicant must use the following analysis for every capsule removed and tested.

During the plant's life the applicant must recalculate the pressure-temperature operating limits based on the greater of the following:

(1) the actual shift in reference temperature for plate C7151-1 and weld 1NMM/KN203/0171 as determined by impact testing, or

(2) the predicted shift in reference temperature for weld 1NMM/S-3986/3876 and plate C 7158-1 as determined by Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

Although material from the most limiting weld seam and plate are not contained in the Zimmer Unit 1 materials surveillance program, we have found that an exemption to Paragraph II.B of Appendix H, 10 CFR Part 50, is justified because methods of analysis contained in Regulatory Guide 1.99, which will be used to determine the radiation induced change in fracture toughness of limiting beltline weld and plates, are conservative.

The applicant has stated that the CVN impact test surveillance specimens are oriented in the longitudinal direction in accordance with the ASME Code in effect at that time. Therefore, the applicant cannot comply with the specimen orientation requirements of ASTM E 185-73. However, we conclude that the test specimens with longitudinal orientation will provide sufficient data to predict the relative change in RT_{NDT} due to neutron irradiation. Our conclusion is based on previously obtained test data and experience that indicate that the relative shift in RT_{NDT} is not significantly sensitive to specimen orientation. Based on our evaluation, we conclude that an exemption to the specimen orientation requirements of Paragraph II.B is justified because equivalent measures of irradiation damage can be obtained from the longitudinal oriented specimens.

The applicant has indicated that for two of the three surveillance capsules there are 24 CVN impact test specimens which are divided equally among weld metal, heat affected zone, and base metal samples. Based on past experiences in making CVN impact temperature curves, we believe that eight specimens are enough to generate the CVN impact temperature curves which are required to determine the adjusted reference temperature.

For the above reasons we conclude that an exemption to paragraph II.B of Appendix H, 10 CFR Part 50, which requires the surveillance program comply with ASTM E 185-73, is justified.

Conclusions for Compliance to Appendices G and H, 10 CFR Part 50

Our technical evaluation has not identified any practical methods by which the existing Zimmer Unit 1 reactor vessel can comply with the specific requirements of Paragraphs III.B.3, III.B.4, III.C.2, IV.A.1, IV.A.3, and IV.B of Appendix G and Paragraph II.B of Appendix H, 10 CFR Part 50. However, the alternate methods proposed to demonstrate compliance with these paragraphs of Appendices G and H have been reviewed and evaluated, and have been found to demonstrate that the safety margins required by Appendices G and H have been achieved.

Compliance with Appendices G and H and the fracture toughness requirements of Section III of the ASME Code ensures that the ferritic components in the primary coolant pressure boundary will behave in a nonbrittle manner, that the probability of rapidly propagating fracture is minimized and that an appropriate material surveillance program exists to monitor radiation damage for the reactor pressure boundary. Compliance with the requirements of the NRC regulations and the specified codes and standards satisfies the requirements of the Commission's General Design Criteria 31 and 32.

Based on the foregoing, pursuant to 10 CFR, Section 50.12, exemptions from the specific requirements of Appendices G and H of 10 CFR Part 50, as discussed above, are authorized by law and can be granted without endangering life or property or the common defense and security and are otherwise in the public interest. We conclude that the public is served by not imposing certain provisions of Appendices G and H of 10 CFR Part 50 that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Furthermore, we have determined that the granting of these exemptions does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. We have concluded that these exemptions would be insignificant from the standpoint of environmental impact statement, and pursuant to 10 CFR 51.5(d)(4) that an environmental impact appraisal need not be granted in connection with this action.

5.3.2 Pressure-Temperature Limits

Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Materials Surveillance Program Requirements," 10 CFR Part 50, describe the conditions that require pressure-temperature limits for the reactor coolant pressure boundary and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins for the reactor coolant pressure boundary at least as great as the safety margins recommended in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, "Protection Against Non-Ductile Failure." Appendix G, 10 CFR Part 50, requires additional safety margins whenever the reactor core is critical, except for low-level physics tests.

The following pressure-temperature limits imposed on the reactor coolant pressure boundary during operation and tests are reviewed to ensure that they provide adequate safety margins against non-ductile behavior or rapidly propagating failure of ferritic components as required by General Design Criterion 31:

(1) Preservice hydrostatic tests,

(2) Inservice leak and hydrostatic tests,

(3) Heatup and cooldown operations, and

(4) Core operation.

Appendices G and H, 10 CFR Part 50, require the applicant to predict the shift in reference temperature due to neutron irradiation. The shift in RT_{NDT} due to neutron irradiation is then added to the initial RT_{NDT} to establish the adjusted reference temperature. The base plate or weld seam having the highest adjusted reference temperatures is considered the most limiting materials upon which the pressure-temperature operating limits are based. In the case of Zimmer Unit 1, the most limiting material is plate C7158-1. Once in service, the pressure-temperature limits must be revised to reflect the actual neutron radiation damage as determined from the results of the reactor vessel materials surveillance program.

The applicant lowest service temperatue is 98°F which is below the lowest service temperature for the MSIV as discussed in SER Section 5.3.1. However, the lowest service temperature of the MSIV will not affect the pressure-temperature limits for the reactor vessel because significant pressure is not applied to the main steam isolation valves until the boiling point of water, viz., 212°F.

According to our evaluation the proposed heatup and cooldown pressure temperature limits (FSAR Figures 16.3-7A and 16.3-7B) are acceptable until the first refueling. After the first refueling the applicant will verify the predicted neutron fluence by dosimetry measurements. This dosimetry measurement will then be utilized to predict the neutron fluence for calculating the pressure temperature limit curves subsequent to the first fuel reloading, and prior to removal of the first surveillance capsule. The calculated shift in RT_{NDT} for the reactor vessel belting must be based on Regulatory Guide 1.99. After removal of the first surveillance capsule the applicant must recalculate the pressure temperature limit curves based on the analysis discussed in SER Section 5.3.1.

The pressure-temperature limits to be imposed on the reactor coolant system for all normal operating, testing and anticipated transient conditions, to ensure adequate safety margins against nonductile or rapidly propagating failure, are in conformance with established criteria, codes, and standards acceptable to the staff. The use of the operating limits based on these criteria, as defined by applicable regulations, codes, and standards, provides reasonable assurance that nonductile or rapidly propagating failure will not occur, and constitutes an acceptable basis for satisfying the applicable requirements of General Design Criterion 31.

5.3.3 Reactor Vessel Integrity

We have reviewed the FSAR sections related to the reactor vessel integrity of Zimmer Unit 1. Although most areas are reviewed separately in accordance with other review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted.

We have reviewed the information in each area to ensure that it is complete and that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed are:

1. Design (SER § 5.3.1)

2. Materials of construction (SER § 5.3.1)

3. Fabrication methods (SER §5.3.1)

4. Operating conditions (SER §5.3.2)

We have reviewed the above factors contributing to the structural integrity of the reactor vessel and conclude that the applicant has complied with Appendices G and H, 10 CFR Part 50, except for Paragraphs III.B.3, III.B.4, III.C.2, IV.A.1, IV.A.3, and IV.B of Appendix G, and Paragraph II.B of Appendix H, for which the applicant has provided sufficient information to justify exemptions.

Paragraph III.B.3 of Appendix G requires that the temperature instruments and Charpy test machines be calibrated per Paragraph NB-2360 of the ASME Code. The standard practice of the time and past experience with Charpy testing make it unlikely that the test instruments were not adequately calibrated and that an exemption to Paragraph III.B.3 is justified.

Paragraph III.B.4 of Appendix G requires the applicant to conduct impact testing according to written procedures. Although the tests were not conducted to formal written procedures for Zimmer Unit 1 impact tests, the applicant has supplied sufficient information to demonstrate that the tests were conducted correctly, and therefore, we have concluded that an exemption to Paragraph III.B.4 is justified.

Paragraph III.C.2 of Appendix G requires that the base metal used to prepare test specimens be taken from excess base metal from the vessel beltline region. The weld specimens for testing were not prepared from excess production plate. The applicant, however, has supplied sufficient data to demonstrate that the weld specimens do represent the welds in the vessel beltline region. Therefore, an exemption to Paragraph III.C.2 is justified.

Paragraph IV.A.1 of Appendix G requires that a reference temperature, RT_{NDT}, be determined per Paragraph NB 2330 of the ASME Code for each ferritic material in the reactor coolant pressure boundary. Although the applicant did not determine the RT_{NDT} per Paragraph NB 2330 of the ASME Code for each ferritic material, the critical RT_{NDT} for operating, maintenance, and testing conditions has been deter[illegible]ed based on additional information available in the literature and additional [illegible]ata supplied by the applicant. Therefore, we have concluded that an exempt[illegible]n to paragraph IV.A.1 of Appendix G is justified.

Paragraph I[illegible].A.3 requires, in part, that the materials for valves meet CVN impact requirements in paragraph NB-2330 of the ASME Code. Although the applicant has not CVN impact tested the MSIV materials, the applicant has supplied sufficient data from other similar material[illegible] to demonstrate that the MSIV would meet the CVN impact requirements of Paragraph NB 2330 of the ASME Code and, therefore, an exemption to Paragraph IV.A.3 is justified.

Paragraph IV.B requires that the reactor vessel beltline materials have a minimum upper-shelf CVN energy of 75 ft-lb unless it can be demonstrated that lower values of upper-shelf CVN energy still provide adequate margin for irradiation deterioration. Although the applicant had not tested all reactor vessel beltline material over a sufficient temperature range to determine whether each material has a minimum upper shelf energy of 75 ft lbs, the applicant has supplied sufficient information from other plants to demonstrate that the CVN impact upper shelf energeis for the Zimmer Unit 1 reactor vessel beltline materials exceed 75 ft lbs. Therefore, we conclude than an exemption to paragraph IV.B is justified.

Paragraph II.B, Appendix H, requires that the material surveillance program comply with ASTM E 185-73. The materials in Zimmer Unit 1's surveillance program does not comply with all requirements in ASTM E 185; however, the materials that are in the program, together with methods for predicting radiation damage provide sufficient information for us to have concluded that an exemption to Paragraph II.B, Appendix H, is justified.

We have reviewed all factors contributing to the structural integrity of the reactor vessel and conclude there are no special considerations that make it necessary to consider potential reactor vessel failure for Zimmer Unit 1.

5.4 Component and Subsystem Design

5.4.1 Reactor Core Isolation Cooling System

The reactor core isolation cooling system (RCIC) has controls which can shut down the system if the ambient temperature in the equipment room exceeds the assigned limits. We requested the applicant to show that the system will not be shut down due to spurious temperature signals. The applicant's response states that the trip point will be established by calculating a heat balance for the normal room environment, and then introducing the heat release caused by alarm limit leak. Actual ambient temperature will be determined during start-up testing. This is acceptable to us. Spurious isolations due to high initial steam flow in the turbine steam lines is discussed under item II.K.3.15 of NUREG-0737 in Section 22 of this SSER.

To protect the reactor core isolation cooling system pump against the effects of water hammer when starting, a jockey pump keeps the system full. An alarm is sounded if the pump stops. To assure that the piping is filled with water, we require by technical specification periodic (every 31 days) high point venting of RCIC discharge piping.

Normally the reactor core isolation cooling system is connected to a nonseismic Category I water source. However, the system is designed to initiate an automatic switchover to the suppression pool during low water level in the condensate storage tank. This automatic feature ensures a water supply for RCIC in the event of a safe shutdown earthquake.

We conclude that the design of the RCIC system conforms to our regulations and applicable Regulatory Guides and is acceptable.

5.4.2 Residual Heat Removal System

There is a single line from the recirculation system to the residual heat removal system for use in cooling the reactor in the shutdown mode. This line is vulnerable to a single failure of either of the isolation valves. The applicant has an alternate cooling path using the safety-relief valves and suppression pool cooling in the event of a failure in the suction line which would preclude residual heat removal system operation. Both paths are operable from emergency power. At our request, the applicant has provided a long term air supply to the automatic depressurization system valves by inclusion of a seismic Category I backup system consisting of two banks of nitrogen cylinders with associated valves and piping. This system provides sufficient capacity (over 10 days) to all six automatic depressurization system valves. Spare cylinders will be maintained onsite to further increase the capacity. We find that sufficient provisions exist to achieve cold shutdown by alternate methods.

To demonstrate valve operability to provide adequate fluid relief for the shutdown cooling mode of operation, the applicant has committed to participate in a BWR Owners Group Test Program which will test this capability for Crosby valves similar to those used in Zimmer. The applicant has also provided analyses which indicate that three of the six ADS valves are required to pass the flow required to achieve cold shutdown. On this basis, but subject to any consideration which might arise as a result of the Owner's Group Test program, we find this capability acceptable.

We conclude that the design of the residual heat removal system conforms to the Commission's regulations and to the applicable regulatory guides and is, therefore, acceptable.

5.4.3 Reactor Water Cleanup System

Evaluation Finding

NUREG-0528 states that the Zimmer reactor water cleanup system will ensure operation within the limits defined in Regulatory Guide 1.56 "Maintenance of Water Purity In Boiling Water Reactors." Since the staff reached that conclusion Regulatory Guide 1.56 has been revised and, therefore, the Zimmer water chemistry was rereviewed against Regulatory Guide 1.56, Revision 1.

Reactor Water Cleanup System Maintenance of Water Purity in Boiling Water Reactors

The reactor water cleanup system continuously removes solid and dissolved impurities from the reactor water through filter demineralizers. The filter demineralizers are pressure precoat type using filter aid and finely ground mixed ion-exchange resins as a filter and ion-exchange medium. The limits of the conductivity, pH, and chloride concentration in the reactor water have been established in the Technical Specification in accordance with the recommendations of Regulatory Guide 1.56, Revision 1 (July 1978). The conductivity and pH will be continuously monitored prior to startup, during power operation, hot standby, and cold shutdown, to ensure that their limits will not be exceeded. High conductivity will be annunciated in the control room. Surveillance requirements and limiting conditions for operation are specified in the proposed Zimmer plant Technical Specifications in accordance with the Standard Technical Specifications. The appropriate corrective actions will be taken when the limits of the conductivity, pH, or chloride concentration in the reactor coolant are exceeded.

We determined that the reactor water cleaning program meets (1) the regulatory positions of Regulatory Guide 1.56, revision 1 (July 1978), (2) the water purity acceptance criteria 2.1 and 4.1 of the Standard Review Plan, Section 5.4.8, and (3) the requirements of General Design Criterion 14 of Appendix A to 10 CFR Part 50, as related to maintaining water purity. On this basis, we conclude that the applicant's program for maintaining the primary coolant water purity by the reactor water cleanup system is acceptable.

Condensate Cleanup System

The condensate cleanup system consists of six deep-bed type (mixed resin) demineralizers to remove solid and dissolved impurities from the condensate of the main condenser to ensure the supply of high purity water to the reactor. The sixth demineralizer, which is a standby demineralizer, will be placed into service to replace an inservice unit at the end of its service run. Each demineralizer has an effluent resin strainer to prevent resin carryover. The limits for the conductivity, chloride concentration, pH and dissolved suspended solids in the demineralizer effluent during power operation have been established and will be implemented by plant operating procedures. The conductivity is continuously monitored for the system influent and effluent and each demineralizer bed effluent. Sample line valves are provided in each demineralizer effluent line and the influent and effluent headers to permit analysis of the water quality.

The applicant has proposed an alternate approach to the guideline of Regulatory Guide 1.56 to regenerate demineralizer resin at least semiannually. The applicant will first regenerate the bed with the lowest remaining anion capacity which will not be permitted to decrease below one-half of the total anion capacity. The expended capacity and the remaining anion capacity is computed at least weekly based on inlet conductivity and integrated flow. The condensate flow rates through each demineralizer will be recorded and integrated on individual counters. We find this alternate approach acceptable. The capacity of new resins will be measured in accordance with ASTM Standard D2187-77. "Standard Methods of Test of Physical and Chemical Properties of Ion-Exchanger Resins." Individual high demineralizer effluent conductivity will be alarmed at the control panel. Conductivity meter alarm setpoints will be set in accordance with the recommendations of Regulatory Guide 1.56, Revision 1 (July 1978). The condensate cleanup system is designed to operate in a manner such that corrective action is initiated prior to exceeding the lower limits of Table 2 of Regulatory Guide 1.56, Revision 1.

We determined that the condensate cleanup system meets (1) the regulatory positions of Regulatory Guide 1.56, Revision 1 (July 1978), (2) the water purity acceptance criterion 1 of Section 10.4.6 of the Standard Review Plan, and (3) the requirements of Criterion 14 of Appendix A to 10 CFR Part 50, as it relates to water chemistry control. On this basis, we conclude that the applicant's condensate cleanup system is acceptable.

6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Engineered Safety Features Metallic Materials

Containment Pressure Boundary Fracture Toughness

The fracture toughness of the ferritic materials that constitute the containment pressure boundary of the Zimmer nuclear plant was reviewed to assess compliance with Criterion 51 of the General Design Criteria (GDC-51), "Fracture Prevention of Containment Pressure Boundary." The Zimmer primary containment (the drywell) is a load-bearing reinforced concrete structure with a thin steel liner on the inside surface which is designed to serve as a membrane providing leak tightness and is capped with a steel head (drywell head) which is not backed with concrete. The fracture toughness requirements of GDC-51 apply to those ferritic steel parts of the containment pressure boundary which are not supported by concrete and are thus load-bearing. These materials are typically applied in containment penetrations such as the equipment hatch, personnel hatch, and pipe system penetrations and, in the case of the Zimmer primary containment, for the drywell head.

The applicant has stated in the FSAR that the ASME Code Section III, 1971 Edition, through the Winter 1971 Addenda was applied in the fabrication of the containment.

Compliance with the requirements of the ASME Code for the ferritic steel parts of the containment pressure boundary satisfies the requirements of GDC-51.

Fracture Prevention of Containment Pressure Boundary

We have assessed the ferritic materials in the Wm. H. Zimmer Nuclear Power Station Unit 1 containment system that constitute the containment pressure boundary to determine if the material fracture toughness is in compliance with the requirements of General Design Criterion 51, "Fracture Prevention of Containment Pressure Boundary."

GDC-51 requires that under operating, maintenance, testing, and postulated accident conditions (1) the ferritic materials of the containment pressure boundary behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The Zimmer Unit 1 primary containment is a reinforced concrete structure with a thin steel liner on the inside surface which serves as a leaktight membrane. The ferritic materials of the containment pressure boundary which were considered in our assessment were those applied in the fabrication of the equipment hatch, personnel airlocks, penetrations, drywell head and piping system components, including the isolation valves required to isolate the system. These components are the parts of the containment system which are not backed by concrete and must sustain loads.

The Zimmer Unit 1 containment pressure boundary is comprised of ASME Code Class 1, 2, and MC components. In late 1979, we reviewed the fracture toughness requirements of the ferritic materials of Class MC, Class 2, and Class 1 components which typically constitute the containment pressure boundary. Based on this review, we determined that the fracture toughness requirements contained in ASME Code Editions and Addenda typical of those used in the design of the Zimmer Unit 1 primary containment may not ensure compliance with GDC-51 for all areas of the containment pressure boundary. We initiated a program to rev... fracture toughness requirements for containment pressure boundary materials for the purpose of defining those fracture toughness criteria that most appropriately address the requirements of GDC-51. Prior to completion of this study, we have elected to apply in our licensing reviews the criteria identified in the Summer 1977 Addenda of Section III of the ASME Code for Class 2 components. These criteria were selected to ensure that uniform fracture toughness requirements, consistent with the containment safety function, are applied to all components in the containment pressure boundary. Accordingly, we have reviewed the Class 1, 2, and MC components in the Zimmer Unit 1 containment pressure boundary according to the fracture toughness requirements of the Summer 1977 Addenda of Section III for Class 2 components.

Our assessment of the fracture toughness of the materials of the Zimmer Unit 1 containment pressure boundary is based on fracture toughness data provided by the applicant and on correlations of the metallurgical characterization of the materials with fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pipe Supports", USNRC, October 1979 and ASME Code Section III, Summer 1977 Addenda, Subsection NC.

The metallurgical characterization of these materials, with respect of their fracture toughness, was developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication.

The metallurgical characterization of these materials, with respect of their fracture toughness, was developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterizations of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of the ASME Code Section III, Subsection NC, provided, in part, the technical basis for our revaluation of compliance with Code requirements.

Based on our review of the available fracture toughness data and material fabrication histories, and the use of correlations between metallurgical characteristics and material fracture toughness, we conclude that the ferritic components in the Zimmer Unit 1 containment pressure boundary meet the fracture toughness requirements that are specified for Class 2 components by the Summer 1977 Addenda of Section III of the ASME Code. We conclude that compliance with these Code requirements provides reasonble assurance that the materials of the Zimmer Unit containment pressure boundary will behave in a non-brittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of GDC 51 are met.

6.2 Containment Systems

6.2.1 General

Steam Bypass of the Suppression Pool

As stated in NUREG-0528, we requested the applicant to provide information to establish the availability of the wetwell spray system 10 minutes following a loss-of-coolant accident. The consequences of actuation of the wetwell spray system on the emergency core cooling system function were to be evaluated also.

The applicant has provided us with the necessary information and analyses and our evaluation is contained in Subsections 6.3.4 and 7.3.3.

6.2.2 Loss of Coolant Accident and Safety Relief Valve Discharge Pool Dynamics

In this Zimmer Safety Evaluation Report "NUREG-0528", dated January 1979, we stated that our review of the Mark II pool dynamics loads are being conducted under two generic technical activities (Task A-8, "Mark II Containment Pool Dynamic Loads"; and Task A-39, "Determination of Safety Relief Valves (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments"). This supplement presents our evaluation of the proposed load specifications for the Zimmer Nuclear Power Station, Unit 1, relative to the generic acceptance criteria developed within the above mentioned activities.

In October 1978, we issued a report, NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," to address the portion of the Mark II Owners Group's program that provides a generic methodology for establishing design basis loss-of-coolant accident and safety/relief valve loads for the lead Mark II plants (La Salle, Zimmer, Shoreham). The load evaluations were conducted by us and our consultants at the Brookhaven National Laboratory.

Since the issuance of NUREG-0487, the Mark II owners submitted additional reports in which they proposed alternative load methodologies for use in the evaluation of Mark II plants. We and our consultants studied these alternative methodologies proposed by the Mark II Owners Group. As a result of these reports, we issued Supplement 1 to NUREG-0487, on September 1980. This supplement contains an evaluation of the proposed alternatives to the lead plant acceptance criteria.

In addition, the Mark II Owners Group conducted additional tests in a modified 4T facility (referred to as 4TCO) to answer questions raised by us regarding the influence of vent length effects on the condensation oscillation loads. These loads are the result of condensing of steam exiting the vent lines. The results of these tests indicate that the condensation oscillation and chugging load specifications set forth in NUREG-0487 for the lead Mark II plants need to be modified based on conservative interpretation of the new 4T data. The Mark II Owners Group proposed new condensation oscillation and chugging load specifications based on data obtained from the modified 4T facility. The new load specifications were submitted on the La Salle docket in July 1980. We and our consultants have completed our review of the new condensation oscillation and chugging load specifications and found them to be acceptable in Supplement 2 to NUREG-0487 which was issued in February 1981.

A summary of our review status for each of the pool dynamic loads is presented in Table 6.1. This table provides a description of each load or phenomenon, the Mark II Owners Group's load specification; it also references our review status and the applicant's position on each load.

As indicated in Table 6.1, the applicant agreed to adopt all but three of our generic criteria. These items relate to quencher air clearing loads (Load II.B in Table 6.1) condensation oscillation and chugging loads (Loads I.C 2.a, b and c in Table 6.1). Alternative criteria were proposed by the applicant for these items. Our evaluation of these alternative criteria is provided below.

Quencher Air Clearing Load (Load II.B. in Table 6.1)

The applicant has committed to install a T-quencher device designed by the Kraftwerk Union of Germany in lieu of the ramshead end devices as originally designed. At the time NUREG-0487 was issued, detailed performance data for the T-quencher device were not available. Therefore, we specified that the load methodology developed for ramshead devices should be employed to compute the bubble pressure amplitude (Criterion II.2.C, Appendix D of NUREG-0487). Subsequent to the issuance of NUREG-0487 and in view of the availability of new test data for the T-quencher. the lead plant applicants proposed an alternative to our acceptance criteria. We issued our evaluation of the alternative acceptance criteria in Supplement 1 to NUREG-0487, dated September 1980. The Zimmer applicant indicated that they would adopt our acceptance criteria for the T-quencher except for the criterion on frequency range. It was determined by the applicant that a frequency range of 3.4 to 10 Hertz as used for Zimmer is considered adequate and conservative. We find this to be acceptable based on calculations by the applicant that show the deviation of the spectral input has negligible effect on the total response contributed by all modes and that conservative assumptions of simultaneous occurrence of safe shutdown earthquake, loss-of-coolant accident and safety/relief valve events were used in the design.

Although the T-quencher spectra peaks are lower in amplitude than the ramshead spectra peaks, the frequency of the peaks shifts in the low frequency direction may cause a small increase in the required number of piping restraints for low-frequency systems. To confirm the adequacy of the Zimmer design, the applicant performed spectral analyses for typical building and piping responses using both ramshead and T-quencher methodologies and concluded that the design can accommodate the T-quencher loads.

Based on the discussion outlined above, we conclude that the current ramshead design basis is an acceptable design basis for Zimmer.

Interim Condensation Oscillation and Chugging Loads (Loads I.C.2.a, I.C.2.b and I.C.2.c in Table 6.1)

As stated earlier, following the issuance of NUREG-0487 in October 1978, the staff expressed concerns regarding the steam condensation load specifications. To avoid impact on the schedule for operation, the Zimmer applicant implemented a program, based on conservative empirical load design, that required a significant amount of design work and plant modifications to ensure high margin of safety and to accommodate any future changes in pool dynamic loads that might occur as a result of the staff's continued review of the new 4TCO data. This program was submitted to the staff in December 1979.

TABLE 6.1
CONFORMANCE OF ZIMMER DESIGN TO NRC ACCEPTANCE CRITERIA

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS*	ZIMMER POSITION ON ACCEPTANCE CRITERIA
I. LOCA-Related Hydrodynamic Loads			
A. Submerged Boundary Loads During Vent Clearing	24 psi over-pressure added to local hydrostatic below vent exit (walls and basemat) - linear attenuation to pool surface.	Acceptable [2]	Acceptable.
B. Pool Swell Loads			
1. Pool Swell Analytical Model			
a) Air Bubble Pressure	Calculated by the pool swell analytical model (PSAM) used in calculation of submerged boundary load.	Acceptable [2]	Acceptable.
b) Pool Swell Elevation	Use PSAM with polytropic exponent of 1.2 to a maximum swell height which is the greater of 1.5 vent submergence or the elevation corresponding to the drywell floor uplift ΔP per NUREG 0487 criteria I.A.4. The associated maximum wetwell air compression is used for design assessment.	Acceptable [2]	Acceptable.
c) Pool Swell Velocity	Velocity history vs. pool elevation predicted by the PSAM used to compute impact loading on small structures and drag on gratings between initial pool surface and maximum pool elevation and steady-state drag between vent exit and maximum	NRC Criteri I.A.2 [1]	Acceptable. The impact of a 10 percent increase in pool swell velocity has been assessed and concluded that the design is adequate.

* See notes at end of table

TABLE 6.1 (Cont'd)
CONFORMANCE OF ZIMMER DESIGN TO NRC ACCEPTANCE CRITERIA

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS*	ZIMMER POSITION ON ACCEPTANCE CRITERIA
	elevation. Analytical velocity variation used up to maximum velocity. Maximum velocity applies thereafter up to maximum pool swell.		
d) Pool Swell Acceleration	Acceleration predicted by the PSAM. Pool acceleration is utilized in the calculation of acceleration drag loads on submerged components during pool swell.	Acceptable. [1]	Acceptable.
e) Wetwell Air Compression	Wetwell Air Compression is calculated by PSAM.	NRC Criteria II.A.2[2]	Acceptable.
f) Drywell Pressure History	Plant unique. Utilized in PSAM to calculate pool swell loads.	Acceptable if based on NEDM-10320. Other wise plant unique reviews required. [1]	Acceptable.
2. Loads on Submerged Boundaries	Maximum bubble pressure predicted by the PSAM added uniformly to local hydrostatic below vent exit (walls and basemat) linear attenuation to pool surface. Applied to walls up to maximum pool swell elevation.	Acceptable. [1]	Acceptable.
3. Impact Loads			
a) Small Structures	1.5 x Pressure-Velocity correlation for pipes and I beams. Constant duration pulse.	NRC criteria I.A.6[1]	Acceptable.

TABLE 6.1 (Cont'd)
CONFORMANCE OF ZIMMER DESIGN TO NRC ACCEPTANCE CRITERIA

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS*	ZIMMER POSITION ON ACCEPTANCE CRITERIA
b) Large Structures	None - Plant unique load where applicable.	Plant unique review.	Acceptable. Zimmer has no large structures in the pool swell zone.
c) Grating	No impact load specified. P_{drag} vs. open area correlation and velocity vs. elevation history from the PSAM.	NRC Criteria I.A.3 [1]	Acceptable. Zimmer has no grating in pool swell area.
4. Wetwell Air Compression			
a) Wall Loads	Direct application of the PSAM calculated pressure due to wetwell compression.	Acceptable.	Acceptable.
b) Diaphragm Upward Loads.	2.5 psid.	NRC Criteria I.A.4 [1]	Acceptable.
5. Asymmetric LOCA Pool Boundary Loads	Use 10 percent of maximum bubble pressure statistically applied to 1/2 of the submerged boundary.	NRC Criteria in Section II.A.3 [2]	Acceptable.
C. Steam Condendation and Chugging Loads			
1. Downcomer Lateral Loads			
a) Single Vent Loads	8.8 KIP static	NRC Criteria I.B.1 [1]	Acceptable.
b) Multiple Vent Loads	Prescribes variation of load per downcomer vs. number of downcomers.	NRC Criteria I.B.2 [1]	Acceptable.

TABLE 6.1 (Cont'd)
CONFORMANCE OF ZIMMER DESIGN TO NRC ACCEPTANCE CRITERIA

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS*	ZIMMER POSITION ON ACCEPTANCE CRITERIA
2. Submerged Boundary Loads			
a) High Steam Flux Loads	Interim load specification submitted on LSCS docket [5]	Load specification and staff's evaluation is presented in [4].	Addressed in this report.
b) Medium Steam Flux	Interim load specification submitted on LSCS docket [5]	Load Specification and Staff's evaluation is presented in [4]	Addressed in this report.
c) Chugging Loads	Representative pressure fluctuation taken from 4TCO test added local hydrostatic.	Load specification and staff's evaluation is presented in [4].	Addressed in this report.
- uniform loading conditions	Interim load specification submitted on the LSCS docket [5]	Load specification and staff's evaluation is presented in [4].	Addressed in this report.
- asymmetric loading condition	Maximum amplitude uniform below vent exit; linear attenuation to pool surface. 20 psi maximum overpressure; -14 psi maximum underpressure and 20-30 Hz frequencies. Peripheral variation of amplitude follows observed statistical distribution with maximum and minimum diametrically opposed.	Acceptable.	Acceptable.

TABLE 6.1 (Cont'd)
CONFORMANCE OF ZIMMER DESIGN TO NRC ACCEPTANCE CRITERIA

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS*	ZIMMER POSITION ON ACCEPTANCE CRITERIA
II. SRV-Related Hydrodynamic Loads			
A. Pool Temperature Limits for KWU and GE four arm quencher.	None specified.	NRC Criteria II.1 and II.3 [1]	Acceptable.
In-plant SRV test	Plant Unique.	Addressed in this report.	Commit to perform SRV in-plant test.
B. Quencher Air Clearing	SSES method is used for T-quencher load definition. [3]	NRC Criteria in Section II.B.5 [2]	Acceptable except for quencher frequency. 3.4-10 Hz used for Zimmer is considered adequate and conservative. (The staff finds this approach acceptable. In-plant tests will be run to demonstrate the adequacy and conservatism of the design loads.
C. Quencher Tie-Down Loads			
1. Quencher Arm Loads			
a) Four Arm Quencher	Vertical and lateral arm loads developed on the basis of bounding assumptions for air/water discharge from the quencher and conservative combinations of maximum/minimum bubble pressure acting on the quencher.	Acceptable.	Acceptable.

TABLE 6.1 (Cont'd)

CONFORMANCE OF ZIMMER DESIGN TO NRC ACCEPTANCE CRITERIA

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS*	ZIMMER POSITION ON ACCEPTANCE CRITERIA
b) KWU T-quencher	T-quencher arm loads specified for SSES. [3]	Review Continuing	Acceptable. These loads will be calculated using the methodology and assumptions described in DFFR for four arm quenchers, as recommended in the Acceptance Criteria. The KWU T-quencher methodology was used to verify the conservatism of this approach.
2. Quencher Tie-Down Loads			
a) Four-Arm Quencher	Includes vertical and lateral arm load transmitted to the basemat via the tie downs. See II.C.1.a above plus vertical transient wave and thrust loads. Thrust load calculated using a standard momentum balance. Vertical and lateral moments for air or water clearing are calculated based on conservative clearing assumptions.	Acceptable.	Acceptable.
b) KWU T-quencher	T-quencher tie-down loads specified for SSES. [3]	Review Continuing	Acceptable. These loads will be calculated using the methodology and assumptions described in DFFR for four arm quenchers. The KWU T-quencher methodology was used to verify the conservatism of this approach.

TABLE 6.1 (Cont'd)
CONFORMANCE OF ZIMMER DESIGN TO NRC ACCEPTANCE CRITERIA

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS*	ZIMMER POSITION ON ACCEPTANCE CRITERIA
III. LOCA/SRV Submerged Structure Loads			
A. LOCA/SRV Jet Loads			
1. LOCA/Ramshead SRV Jet Loads	Methodology based on a quasi-one-dimensional model.	NRC Criteria [2]	Acceptable.
2. SRV-Quencher Jet Loads	No loads specified for lead plants. Model under development in long-term program.	NRC Criteria [2]	Acceptable.
B. LOCA/SRV Air Bubble Drag Loads			
1. LOCA Air Bubble Loads	Details of methodology are included in Zimmr Appendix G to the FSAR Design Assessment. Report.	NRC Criteria in Section II.C.2 [2]	Acceptable.
2. SRV-Ramshead Air Bubble Loads	The methodology is based on an analytical model in the bubble charging process including bubble rise and oscillation. Acceleration drag along is considered.	NRC Criteria III.B.2[1]	Acceptable.
3. SRV-Quencher Air Bubble Loads	No quencher drag model provided for lead plants. Lead plants propose interim use of ramshead model (See III.B.2 above).	NRC Criteria III.B.3 [1]	The bubble location and radius had been defined appropriately for T-quenchers. Bubbles are located near the arms. The bubble size is predicted from the line air volume.

TABLE 6.1 (Cont'd)

CONFORMANCE OF ZIMMER DESIGN TO NRC ACCEPTANCE CRITERIA

LOAD OR PHENOMENON	MARK II OWNERS GROUP LOAD SPECIFICATION	NRC REVIEW STATUS*	ZIMMER POSITION ON ACCEPTANCE CRITERIA
C. Steam Condensation Drag Loads	No generic load methodology provided.	Zimmer load specification and NRC review is addressed in this report.	Described in Subsections 5.3.1.3.5 and 5.3.1.3.6 of the DAR.
IV. Secondary Loads			
A. Sonic Wave Load	Negligible Load - none specified	Acceptable.	Acceptable.
B. Compressive Wave Load	Negligible Load - none specified	Acceptable.	Acceptable.
C. Post Swell Wave Load	No generic load provided	Plant unique load specification, addressed in this report.	Addressed in Zimmer Closure report.
D. Seismic Slosh Load	No generic load provided	Plant unique load specification, addressed in this report.	Addressed in Zimmer Closure report.
E. Fallback load on Submerged	Negligible load - none specified.	Acceptable.	Acceptable.
F. Thrust Loads	Momentum balance.	Acceptable	Acceptable.
G. Friction Drag Loads	Standard friction drag calculations.	Acceptable.	Acceptable.
H. Vent Clearing Loads	Negligible Load - none specified	Acceptable.	Acceptable.

NOTES TO TABLE

[1] NRC Acceptance Criteria set forth in NREG-0487.

[2] NRC Acceptance Criteria set forth in Supplement 1 of NUREG-0487.

[3] Susquehanna Steam Electric Station's Design Assessment Report.

[4] NRC Acceptance Criteria is issued in Supplement 2 of NUREG-0487.

[5] La Salle County Station Condensation Oscillation and Chugging Load Specifications (letter from L. DelGeorge to B.J. Youngblood dated July 11, 1980).

In February 1981, we issued supplement 2 to NUREG-0487, to address the interim condensation oscillation and chugging load methodologies for establishing design basis steam condensation loads for the lead Mark II plants. As stated in supplement 2 to NUREG-0487, we found that the interim condensation oscillation and chugging load specifications are conservative and acceptable for use in the evaluation of the lead plants. However, since the loads set forth in supplement 2 to NUREG-0487 were developed for the La Salle plant, we indicated that other Mark II plant owners referencing the data in supplement 2 to NUREG-0487 must provide the following:

(1) Information regarding their maximum plant-specific pool temperature to justify the exclusion of selected high temperature 4TCO data, and

(2) Plant-specific unit cell information for their plant to establish the plant-specific geometry factor(s) that will be applied to the maximum condensation oscillation and chugging loads derived from the 4TCO data.

To assess the capability of the Zimmer power station design to accommodate the interim condensation oscillation and chugging load specifications, we have requested the applicant to provide the plant specific information listed in items 1 and 2 above and to provide an acceleration response spectra for comparison of the design basis loads with the staff approved interim loads.

We have reviewed the applicant's submittal and concluded that:

(1) Excluding data whenever the pool temperature exceeds 140 degrees Fahrenheit is justified since the Zimmer mean pool temperature will not attain more than 140 degrees Fahrenheit with a limitation of 100 degrees Fahrenheit as the initial temperature during the design basis accident.

(2) Applying a constant factor of 0.8 to the 4TCO pressure data to arrive at the plant specific condensation oscillation load and a factor of 0.8 to the selected chugging load pressure traces chosen from the 4TCO data are justified since the Zimmer suppression pool is larger on a per vent basis than the 4TCO test cell.

(3) The Zimmer empirical condensation oscillation and chugging load specifications bound the interim generic load specifications.

Based on the discussion outlined above, we conclude that the empirical condensation oscillation and chugging load specifications are acceptable design bases.

In addition to our generic review of the Mark II pool dynamic loads, we have reviewed a limited number of pool dynamic loads on a plant unique basis. The basis of our review of these areas are discussed below.

(1) Drywell Pressure History (Load I.B.1.f first column in Table 6.2)

The drywell pressure history is utilized as part of the overall pool swell load methodology. The applicant has based its calculation of the drywell pressure history on the methods described in General Electric Topical Report NEDO-10320, "The General Electric Pressure Suppression Containment Analytical Model." We previously reviewed this methodology on a generic basis and concluded it was acceptable.

(2) Large Structure Impact Loads (Load I.B.3.b in Table 6.1)

The applicant has stated that the Zimmer facility does not contain any large horizontal structures in the pool swell zone that would be subject to impact loads. Since the applicant has reviewed the as-built plant design and concluded that no large structure exists in the pool swell impact zone, we concur that no load specification is necessary for expansive structures.

(3) Post Swell Wave Load and Seismic Slosh Load (Load IV.C and D in Table 6.1)

These loads have been determined to be secondary loads in that they are not design controlling. We have reviewed the applicant's evaluation of these loads and find them to be acceptable.

(4) Steam Condensation Submerged Drag Loads (Load III.C. in Table 6.1)

Submerged structures in the Zimmer suppression pool were assessed by the applicant for loads due to main vent steam condensation. A procedure was developed to provide a conservative evaluation of these loads. The approach utilizes the same basic approach that was applied to air bubble loads with several modifications. The source strength for these loads was derived from the 4T data. The maximum observed loads on the 4T tank bottom were used to establish bounding source strengths. Randomness in multiple source timing and phasing was accommodated by considering the worst case so as to provide the maximum pressure gradient across a structure. We find this approach conservative and acceptable.

(5) Pool Temperature Limit (Phenomenon II.A in Table 6.1) and Safety Relief Valve In-Plant Test

We require in Criterion II.A of NUREG-0487 that the suppression pool local temperature shall not exceed 200 degrees Fahrenheit for all plant transients involving safety relief valve operations. Since the issuance of NUREG-0487, the Mark II Owners Group proposed alternative suppression pool temperature limits. Our evaluation of the proposed limits, specified below, will be issued in the third quarter of 1981 in NUREG-0783, "Suppression Pool Temperature Limits For BWR Containment."

(1) The suppression pool local temperature shall not exceed 200 degrees Fahrenheit for all plant transients involving safety relief valve operation during which the steam flux through the quencher perforations exceeds 94 pounds per square foot per second.

(2) The suppression pool local subcooling shall not be less than 20 degrees Fahrenheit for all plant transients involving safety relief valve operations during which the steam flux through the quencher perforations does not exceed 42 pounds per square foot per second. This is equivalent to local temperature of 210 degrees Fahrenheit with quencher submergence of 14 feet.

(3) For plant transients involving safety relief valve operations during which the steam flux through the quencher perforations exceeds 42 but is less than 94 pounds per square foot per second, the suppression

pool local temperature can be established by linearly interpreting the local temperature established under item (1) and (2) above.

The above limits are applicable to the Zimmer plant since the applicant is utilizing a quencher device with hole patterns identical to those in the quencher device for which our evaluation was conducted.

The applicant has provided plant unique analyses for pool temperature responses to transients involving safety relief valve operation. Results of the analyses indicate that the plant will operate within the limits specified above. We reviewed the applicant's analyses and concluded that the assumptions used by the applicant are reasonably conservative and in agreement with the staff's recommendations set forth in NUREG-0783 and, therefore, acceptable.

The Zimmer design utilizes 18 local temperature sensors mounted on the pool wall and pedestal wall. The system design provides the operator with necessary information regarding localized heatup during safety relief valve actuation with adequate time to take the necessary action required to assure that the local suppression pool temperature will always remain below the limits specified above. Based on our review of the applicant's proposed pool temperature monitoring system, we conclude that the design meets the NUREG-0487 recommendation and therefore is acceptable.

The applicant has committed to perform a comprehensive safety relief valve in-plant test which is to be completed prior to commercial operation of the facility. These tests will include single and multiple valve tests. The applicant has committed to confirm the adequacy of the piping system design based on the results of these in-plant tests. In addition, the applicant will utilize information from these tests to establish the difference between local and bulk pool temperatures to demonstrate that the maximum local pool temperature specifications will not be exceeded.

In conclusion, we conducted an assessment of the Zimmer facility against our generic acceptance criteria. We also reviewed those few areas where alternative criteria have been proposed. In addition, we completed our review of pool dynamic loads that were relegated to plant unique reviews. In each of these areas, we concluded that the pool dynamic loads utilized by the applicant are conservative and therefore acceptable.

6.2.5 Combustible Gas Control

The accident at Three Mile Island Unit 2 involved a large amount of metal-water reaction in the core with resulting hydrogen generation well in excess of the amounts considered in 10 CFR 50.44 of the Commission's regulations. During the past year the staff has been studying the potential of excess hydrogen generation, the effects such concentrations of hydrogen would have on the various types of plants, and the effectiveness of various mitigation systems in protecting the plant against such situations. The results of our studies to date are presented in the SECY-80-107 series of documents. In these reports, we recommend that all BWR Mark I and II containment plants be inerted and that owners of all other plants be required to provide a proposed design (or designs) to mitigate the consequences of large amounts of hydrogen in containment. The associated proposed interim rule was published in the Federal Register on October 2, 1980.

Subsequent to the issuance of SECY-80-107, a substantial amount of additional work has been performed on this issue with emphasis on ice condensers. With respect to the ice condensers, and specifically Sequoyah, the Commission has decided that the matter of hydrogen control for degraded core accidents in plants with small containments needs to be resolved in the near term, i.e , the resolution should not be deferred to rulemaking.

By letter dated March 16, 1981, the staff stated its position on this matter regarding the Zimmer Station. By letter, dated March 26, 1981, the applicant committed to inerting the Zimmer containment. The inerting system will be inplace and functional by commercial operation of the station.

6.2.6 Containment Leakage Testing

Appendix J Exemption

Appendix J to 10 CFR Part 50 was amended by a final rule effective October 22, 1980 regarding the requirements for the leak testing of the containment building air locks.

The applicant's method for leak testing the containment personnel air locks as outlined in the Zimmer Safety Evaluation Report of January 1979 is now in conformance with the final rule.

6.3 Emergency Core Cooling System

6.3.2 Functional Design

Venting of Emergency Core Cooling System Injection Lines

One of the design requirements of the emergency core cooling system is that cooling water flow be provided rapidly following the initiation signal. By always keeping the emergency core cooling system pump discharge lines full, the lag time between the signal for pump start and the initiation of flow into the reactor pressure vessel can be minimized. In addition, full discharge lines will prevent potentially damaging water hammer occurrences on system startup.

In Zimmer, three jockey pumps are provided to keep the emergency core cooling system injection lines filled with water. Each pump is powered by a different emergency bus so that loss of offsite power coupled with single failure can only disable one fill system. Failure of a jockey pump is alarmed in the control room.

We require periodic high point venting of the emergency core cooling system injection lines to reduce the likelihood of air pocket presence and that this item shall be incorporated in the plant technical specifications.

Long-Term Cooling Capability

The emergency core cooling system pumps must have the capability to operate for an extended period of time during the long term recirculation phase following a loss-of-coolant accident. The applicant states that each RHR pump is designed for a continuous operation of 3 to 6 months. We note that for Zimmer only one RHR pump is required for the longer term and that three are available. Based on the data provided by the applicant and the redundancy of equipment available, we find adequate assurance that the emergency core cooling system pumps will be able to perform the long-term cooling function in the event of an accident.

Isolation Valve Leakage

We noted that the applicant has not addressed long-term leakage from the first isolation valve outside the suppression pool following a loss-of-coolant accident. In a letter dated May 4, 1981 from H. C. Brinkman (C.G.&E) to I. Peltier (NRC), the applicant postulated a leak rate of 5 gallons per minute from the valve (ECCS suction valves are designed with a backseat to prevent leakage through the packing. According to the applicant the expected leakage through the valve seat and packing would not exceed 2 GPM). It was indicated that the leakage will drain to one of the reactor building sumps, each of which is equipped with two 25 gallons per minute pumps. This water can be transferred to the radwaste system. Water processed in the radwaste system can be returned to the condensate storage tank from where it can be injected into the vessel and suppression pool by the High Pressure Core Spray System, the Condensate/Condensate Booster System, the Control Rod Drive Hydraulic System and the Reactor Core Isolation Cooling System. A complete loop is therefore provided to maintain suppression pool inventory.

Adequate time would be available for operator action. Based on a conservative leak rate of 50 gallons per minute leak, at least 40 hours would be required to lower the suppression pool level to 6 inches from normal water level. The operator would be alerted to the leakage by control room alarms activated by level sensors in each of the reactor building sumps and the suppression pool.

The potential for flooding of redundant Emergency Core Cooling System equipment as a result of the leakge was also addressed. The applicant stated that the Emergency Core Cooling System pumps are located in separate cubicles which are watertight.

We find the plant provisions to handle this leakage in the post loss-of-coolant accident period to be acceptable.

6.3.4 Performance Evaluation

Two Loop Test Apparatus Test Results

At the time that NUREG-0528 was issued (January 1979) we reserved judgment with respect to the conformance of the emergency core cooling system performance analysis to 10 CFR 50.46 because the preliminary analysis of the Two Loop Test Apparatus test results indicated a need to investigate further a portion of the General Electric Company emergency core cooling system evaluation model.

Comparison of blowdown tests run in the two loop test apparatus (TLTA) in 1978 raised staff concerns about the conservatism of part of the ECCS evaluation model used by the General Electric (GE) Company. The TLTA configuration is a scaled BWR/6 design and includes the following major components: (1) pressure vessel and internals, (2) an 8 x 8 heated bundle, (3) two recirculation loops, (4) ECC systems (HPCS, LPCS, LPCI), (5) automatic depressurization system, and (6) auxiliary systems.

During August of 1978, test number 6405 was conducted; the test had an average power bundle with low ECC injection flow. Results of the test were compared with those from test 6007 which had the same initial conditions but no ECC injection. The comparison was presented in the monthly report issued in September 1978 and in a program management group meeting on September 21 and 22, 1978. The comparison showed that the system depressurized more slowly with ECC injection than without ECC injection. Since the slower depressurization with ECC injection was contrary to intuitive expectations, GE was requested to discuss the test results and implications with the NRC.

Two theories were advanced as to why ECC injection slows the depressurization: (1) additional steam is produced by ECC fluid contacting the core or hot vessel walls, and (2) increased liquid at the break decreases the volumetric break flow. The first theory led to the concern that the vaporization correlation used to predict steam updraft in the REFLOOD code might underpredict the actual steam updraft and result in a premature breakdown of flooding due to countercurrent flow at the top of the fuel assemblies. Also, if the SAFE code underpredicted the vaporization in the vessel, the calculated depressurization rate would be too high and would result in early prediction of actuation of low pressure ECC systems.

In a letter to W. D. Beckner (NRC) and Dr. M. Merilo (EPRI) from G. W. Burnette (GE), "Further Evaluation and Interpretation of BD/ECC-1A Data," July 31, 1979, GE presented analyses which show that there was increased liquid entrainment in the blowdown flow for the test with ECC injection. Also, the analyses showed that the steam flow through the steam separator above the core was lower with ECC injection than without injection (due to quenching of steam by the spray flow). Therefore, the analysis of the TLTA data shows that the difference in depressurization rate is due to the liquid entrainment in the break flow and not due to increased steaming in the core.

Two repeat tests were conducted in TLTA with (test number 6425) and without (test number 6426) ECC injection. For these repeat tests, improved break flow instrumentation was used to verify that the difference in depressurization rate was due to increased liquid in the break flow rather than increased core steam flow. As discussed in a letter from L. Harold Sullivan (NRC) to Paul S. Check (NRC), "Status Request on Modeling Capabilities of the TLTA Experiment - 6406," February 23, 1981, the repeat tests clearly show that the liquid in the break flow is the reason for the difference in the depressurization rate. Therefore, the concern that the REFLOOD and SAFE codes are underpredicting the steaming rate is without basis, and we find that the GE ECCS evaluation continues to be acceptable.

Low Pressure Coolant Injection Diversion

We stated in NUREG-0528 that we had not completed our review of the proposed automatic diversion of low pressure coolant injection system pumps and procedures relative to diversion.

Low pressure core injection water is diverted automatically (with containment high pressure) after 10 minutes to the wetwell spray in order to increase the allowable suppression pool steam bypass.

At our request, the applicant provided analyses for the worst break loss-of-coolant accident under conditions of low pressure coolant injection diversion. The worst break was found to be a high pressure core spray line break with failure of the LPCS diesel generator which supplies the low pressure core spray pump and Train-A of the low pressure coolant injection system. Train-B of the low pressure coolant injection was diverted leaving only Train-C for core cooling. (LPCI-C is dedicated for core cooling only - there is no heat exchanger in this train). Peak cladding temperature was calculated to be 1725°F which meets the 2200°F limit required by 10 CFR 50.46.

Electrical, instrumentation, and control systems are provided to accomplish automatic diversion of the LPCI system. The wetwell spray will be automatically initiated after a 10 minute period from initiation of a LOCA only if wetwell pressure is $\geq$ 35 psig. The automatic initiation system has a manual override capability, to maintain the system for LPCI operation should wetwell spray operation not be required. The applicant has committed to add the instrumentation used for wetwell spray initiation in plant technical specifications. The acceptability of these systems is discussed in Section 7 of this SSER.

We accept the applicant's LOCA analysis; however, we intend to perform independent audit calculations on a similar plant to confirm the applicant's conclusions. We are planning to perform the independent calculations by the first of FY 82 to provide further confirmation of our conclusion.

Emergency procedures used for LPCI diversion (after 10 minutes) will be reviewed by the staff as part of the TMI-2 requirements (Ref: I.C.1 Short Term Accident and Procedure Review, NUREG-0737).

Residual Heat Removal System (Operator Action)

If a crack in the residual heat removal common suction line outside primary containment is postulated to occur during shutdown cooling, reactor vessel water level would decrease to Level 3 causing isolation of the line. Reactor pressure would rise to the SRV setpoint as a result of the isolation, thus causing safety/relief valves to open and reclose. If all high pressure systems were assumed to be unavailable (plant is shut down), and only low pressure core spray and one low pressure coolant injection were available, manual opening of safety/relief valves would be required to depressurize the vessel. General Electric has evaluated the effects of the crack and analyzed the above scenario for a BWR-5 similar to Zimmer (La Salle) to show how much time the operator has to depressurize the vessel so that low pressure core injection systems can restore the reactor vessel level.

Results from the analysis showed that more than 20 minutes are available for the operator to depressurize the vessel under the postulated conditions. While it is expected that the operator action time would be approximately the same for Zimmer we require the confirmatory analysis to which the applicant has committed (Ref: Letter from J. D. Flynn (C.G.&E.) to H. Denton (NRC) - Response to RSB-17).

Scram System Pipe Break

Since our review of the Zimmer Safety Analysis Report and issuance of our evaluation in NUREG-0528, concerns have been raised on a generic basis for boiling water reactors with regard to the quality of the scram discharge volume piping, the ability to detect and isolate breaks in this system, and the potential for water and steam degradation of available ECCS equipment as a result of break in this system. C.G.&E. has been notified by letter dated April 24, 1981 from R. L. Tedesco (NRC) to E. A. Borgmann (C.G.&E.) that these concerns must be addressed prior to the issuance of an operating license for Zimmer. We will report on this matter in a supplement to this report.

Pending resolution of the issues noted above, we conclude that the emergency core cooling system for Zimmer meets all of the criteria of 10 CFR Part 50.46 and the requirements of Appendix K to 10 CFR Part 50.

6.4 Habitability Systems

6.4.2 Toxic Gas Protection

In our SER of January 1979, we identified chlorine and ammonia as potential toxic gas hazards with respect to the Zimmer Unit 1 control room. We stated that the applicant had committed to provide redundant chlorine and ammonia detectors in the outside intakes of the control room ventilation system. In Revision 68 (December 1980) to the FSAR the applicant stated that these detectors have been provided.

The identification of chlorine and ammonia as potential toxic gas hazards was based on our review of toxic gas hazards within the vicinity of the site. It included the storage of chlorine on site (10 one-ton cylinders in the structure for the circulating water pumps about 500 feet away from the nearest control room outside air intake), the barge traffic of toxic chemicals on the Ohio River (found to be insignificant in accordance with Regulatory Guide 1.78) and the rail traffic on the nearby Chesapeake and Ohio Railroad (found to be of sufficient frequency to warrant design basis toxic gas protection against chlorine and anhydrous ammonia). With respect to truck traffic on U.S. Route 52 near the Zimmer site the staff found the information provided in the FSAR to be insufficient to determine if a toxic gas hazard exists. As stated in Section 2.2 of this supplement, our evaluation of this issue is continuing.

Based on our review we find acceptable the protection provided against chlorine and ammonia gases for the Zimmer Unit 1 control room in accordance with the requirements of General Design Criterion 19. Should we find, as a result of our continuing evaluation as discussed in Section 2.2 of this supplement, that the truck traffic on U.S. Route 52 poses a toxic gas hazard in addition to chlorine and ammonia, we will then reopen the review with respect to toxic gas protection and will require the applicant to provide the approprate additional toxic gas detection and protection capability.

7 INSTRUMENTATION AND CONTROLS

7.1 General Information

7.1.3 Specific Findings

In NUREG-0528 we expressed concern about several "repetitive" problems which appeared in our review of the reactor trip system (subsection 7.2.2), engineered safety features (subsection 7.3.2), systems required for safe shutdown (subsection 7.4.2), safety-related display instrumentation (subsection 7.5.2), all other instrumentation systems required for safety (subsection 7.6.2) and onsite power systems (subsection 8.3.2 and 8.3.3). The "repetitive" problems were characterized as problems with (1) physical separation and isolation, (2) testing of isolation devices, (3) safety system instrument range and set points, (4) safety system instrument response time testing, and (5) seismic qualification of instruments.

Physical Separation of Associated Circuits

The applicant is currently revising the FSAR to clarify the design criteria for associated circuits (non 1E circuits which share power supplies, enclosures or raceways with 1E circuits). This revision will also provide justification for any deviations from Regulatory Guide 1.75, even though this Regulatory Guide did not exist when the Zimmer construction permit was issued. If necessary, we will report on any unacceptable deviations from Regulatory Guide 1.75 in a supplement to this report.

7.1.4 Evaluation

With the exception of the physical separation criteria for associated circuits, which was discussed in Section 7.1.3, we consider the general design of the Zimmer plan acceptable as discussed in our original Safety Evaluation Report.

7.2 Reactor Trip System

7.2.3 Specific Findings

Alternate Reactor Protection System Power Sources

In NUREG-0528, we noted that the alternate reactor protection system power sources resulted in less than ideal physical separation and electrical independence.

The reactor protection system receives normal electrical power from the two motor generator sets. Alternate (backup) power to the reactor protection system in the event a motor generator set should fail is provided from the Division 1 instrument bus 1A (Class 1E power source). As we noted in our SER, several instruments (sensors) were identified which require connection not only to motor generator bus A, but also to either Division 2 or Division 3 direct current circuits (Class 1E). At each of these instruments the wiring from d.c. circuit and the motor generator set were run in the same conduit for several feet. We found this to be unacceptable since the motor generator wiring became "associated" with more than one redundant Class 1E circuit, a violation of Regulatory Guide 1.75. Connecting the motor generator set bus to an alternate Class 1E bus (Division 1 a.c.) compounded the problem since this would also result in redundant Class 1E divisional wiring being run in the same conduit at these instruments (e.g., Division 3 d.c. with Division 1 a.c.).

To correct this design deficiency, the applicant has decided to install additional sensors of the same type on each rack to provide one of the two functions provided by the original instrument. The motor generator set output wiring will be connected to one set of instruments, and the d.c. wiring to the other set, thus separating the (Class 1E) d.c. wiring from the motor generator set output circuit. In addition, connecting the motor generator set bus to an alternate Class 1E a.c. source (thus improving the realiability of the a.c. system) will no longer compromise the independence of any other Class 1E circuits. We agree that the applicant's proposed modification is acceptable.

Backup Scram Capability

As noted in the SER, our review of the wiring for the backup scram system indicated that Division 1 and Division 2 direct current circuits (redundant Class 1E circuits) were routed in the same conduits with each other and with non-Class 1E wiring to the process computer, a clear violation of the separation provisions of Regulatory Guide 1.75. In addition, there was insufficient separation between the Division 1 and 2 wiring at a certain terminal strip. The applicant agreed that these were unacceptable designs and stated that the Division 1 and 2 circuits will be physically separated from each other and from all other wiring, or placed in separate conduit. We believe that the applicant's proposed modification will provide sufficient independence among Class 1E and non-Class 1E circuits discussd above, and is therefore, acceptable.

Cabinet Lighting

In our SER we identified an instance where Class 1E containment isolation wiring was run in proximity to a non-Class 1E lighting circuit, thus causing the lighting circuit to become "associated" with the Class 1E circuit. The applicant agreed that this was an unacceptable design since the lighting circuits are not treated as associated circuits. The applicant has stated that the lighting circuit has been re-routed to eliminate its association with the containment isolation wiring. This modification is acceptable.

Motor Generator Set Protection

On October 31, 1978, General Electric Company submitted a request that we approve the conceptual design of the proposed circuitry for resolving this matter. These modifications were proposed in response to our concern about the capability of the Class IE reactor protection system and other Class IE systems and components powered by the reactor protection system power supplies to accommodate the effects of possible sustained abnormal voltage or frequency conditions from the non-Class IE reactor protection system power supplies. These abnormal conditions could be caused by possible though unlikely combinations of undetectable single failures or by the effects of earthquakes, and could result in damage to the Class IE systems and components powered by the reactor protection system power supplies with the attendant potential loss of capability to perform their intended safety functions.

The proposed modifications to the reactor protection system power supply protective circuitry would consist of the addition of two Class IE "protective packages" in series between each motor-generator set and its respective reactor protection system bus, and between the alternate power source and the reactor protection system buses. Each protective package would include a breaker and associated overvoltage, undervoltage and underfrequency relaying, and would meet the testability requirements for Class IE equipment.

With the protective packages installed, any random undetectable or seismically-induced abnormal voltage or frequency conditions in the outputs of the two motor-generator sets of the alternate power supply would trip either one or both of the two protective packages installed between each power supply and its respective reactor protection system bus thereby producing a half scram on that channel and retaining full scram capability on the other channel. The proposed modifications would provide fully redundant Class IE protection for the Class IE systems and components powered by the reactor protection system power supplies and would thereby bring the overall reactor protection system design into full conformance with Criteria 2 and 21 of the General Design Criteria, Institute of Electrical and Electronic Engineers Standards IEEE-279 and IEEE-379, and the applicable provisions of our Standard Review Plan.

On the bases of its conformance to the aforementioned criteria, we conclude that the conceptual design of the proposed modifications to the reactor protection system power supply protection circuitry is acceptable contingent upon implementation in conformance with the applicable criteria for Class IE systems. We note that the installation of the two protective packages in series with the alternate power source will obviate the need for technical specification time constraints on plant operation while using the alternate power source to supply power to one of the reactor protection system buses.

The applicant committed to implementation of a similar modification of the Zimmer plant. We agree that the applicant's proposed modification is acceptable.

7.2.4 Evaluation

On the basis of the specific findings discussed above (Section 7.2.3) regarding the acceptable resolution of the concerns raised in Section 7.2 of our original Safety Evaluation Report, we consider the Zimmer Reactor Trip System to be acceptable.

7.3 Engineered Safety Feature Systems

7.3.3 Specific Findings

Loss of Safety Function After Reset

As was done for operating reactors through IE Bulletin 80-06, we have requested that the applicant review all safety equipment to determine which, if any, safety functions might be unavailable after reset, and what changes could be implemented to correct any problems. The applicant has not yet replied to this request. We will report on the resolution of this issue in a supplement to this report.

Temperature Monitoring

Power Sources

Two redundant leak detection temperature monitoring systems are provided. One is for the inboard and one is for the outboard isolation valves. Each system has two channels which share calibration equipment. Each system is powered from a separate Class 1E instrument bus. We were concerned that the power supply from both bus 1A and 1B in cabinet 1H13-P642 may compromise the electrical independence and physical separation unless power supply 1E31-K600 is qualified as an isolation device. To address this concern, the applicant has enclosed the cross divisional wiring in conduit and has placed the cross divisional loads in metal cans. We consider this modification acceptable.

Control Room Heating Ventilation and Air Conditioning System

Our safety evaluation report stated that the control room heating ventilation and air conditioning system would be acceptable provided the ammonia detectors were seismically qualified. These detectors are reported in FSAR Section 3.10 to have been successfully qualified. Therefore, the system is now acceptable.

Automatic Initiation of Wet Well Sprays

The applicant has modified the Low Pressure Coolant Injection system so that coolant flow from this system will be diverted to suppress steam pressure surges in the wet well during loss of coolant accidents, provided that adequate coolant is being delivered to the reactor vessel. We have reviewed the logic and initiating circuitry of this modified design and conclude that it is acceptable.

7.3.4 Evaluation

As stated in Section 7.3.3, we will report on the resolution of our concern regarding the loss of safety function after reset in a supplement to this report. Otherwise, based on the acceptable resolution discussed above (Section 7.3.3) of the concerns raised in Section 7.3 of our original Safety Evaluation Report, we consider the Zimmer Engineered Safety Features Systems to be acceptable.

7.4 Systems Required For Safe Shutdown

7.4.3 Specific Findings

In NUREG-0528 we noted that the systems required for safe shutdown had a number of problems associated with physical separation and electrical independence.

7.4.4 Evaluation

(See subsection 7.1.4 for resolution of physical separation and electrical independence issues)

7.5 Safety-Related Display Instrumentation

7.5.3 Specific Findings

Loss of Power to Instruments and Control Systems

We 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 non-safety related instruments and to control systems (This issue was addressed for operating reactors through I&E Bulletin 79-27). The applicant has responded. However, the response did not fully address the concerns. We are pursuing this issue and will report on the resolution of this issue in a supplement to this report.

Emergency Core Cooling

The position of control rods is provided by the rod position indication system, which is a part of the reactor manual control system. In our safety evaluation report (SER) we noted our concern that a failure in Division 2 power coincident with a loss of offsite power could result in a loss of all rod insertion indication. To correct this problem the applicant has transferred the rod position information system power source to the computer uninterruptible power supply. The correct implementation of this modification will be verified by preoperational testing of the reactor Manual Control System. In addition, because the Reactor Manual Control System also serves important functions related to the rod block monitor, the rod sequence control system, and the refueling interlocks, we require that regular surveillance tests of the reactor manual control system be adopted into the Technical Specifications. On these bases, we conclude that the rod position indication system is acceptable.

Regulatory Guide 1.97

We have recently issued Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." This revision reflects a number of major changes in postaccident instrumentation and includes specific implementation requirements for plants in the operating license review stage. Specifically, plants scheduled to be licensed to operate before June 1, 1983, should meet the requirements of NUREG-0737, "Clarification of TMI Action Plan Requirements," and the Commission Memorandum and Order (CLI-80-21) and the schedules of these documents in this area or prior to the issuance of a license to operate, whichever date is later. The balance of the provisions of this guide should be completed by June 1983. We require that the applicant provide its commitment to implement the above stated requirements regarding this regulatory guide.

7.5.4 Evaluation

We will report on the resolution of our concern regarding the loss of power to instruments and control systems discussed in Section 7.5.3 in a supplement to this report. Otherwise, based on the acceptable resolution discussed above (Section 7.5.3) of the concerns raised in Section 7.5 of our original Safety Evaluation Report and the License Condition to meet Regulatory Guide 1.97, Revision 2, we consider the Safety-Related Display Instrumentation of Zimmer to be acceptable.

7.6 All Other Instrumentation Systems Required For Safety

7.6.3 Specific Findings

Source Range Equipment

The source range equipment is similar to previous design. However, in Zimmer the equipment is housed in an interconnected series of instrumentation racks which house all of the other nuclear instrumentation, the rod block monitor, and the process radiation monitoring equipment which initiates reactor building and fuel pool isolation. Furthermore, the relays which provide trip and alarm isolation for the source range, intermediate range and process radiation monitoring equipment are housed in these racks and the safety inputs and isolated alarm relay outputs are bundled together in each of the four auxiliary trip units.

We were originally concerned that this design violated the applicant's separation criteria, but we have subsequently found this design acceptable. The bases for the acceptance of this design are:

(1) The isolation devices are relays. Typical coil voltages are 24 volts dc and typical contact load voltages are 10 vdc. Therefore, the likelihood of there being sufficient energy present to start an electrical fire within the cable bundle is very small.

(2) The relay coils are driven from bistable devices which act as a second level of isolation between the non-Class 1E signals and the nuclear detector analog signal path.

(3) Given that a fault in a non-Class 1E circuit may propagate to multiple Class 1E circuits as the result of fire, the Class 1E circuits which are in this cabinet are not required for safe shutdown and therefore are not required after a fire.

(4) A similar situation (but involving analog to analog isolation) was analyzed at Diablo Canyon several years ago. Several tests were performed which demonstrated that noise in the non-Class 1E circuits would not affect the Class 1E signals. Because of the similarity of the design and function in the Class 1E circuits which are involved, the use of standard grounding techniques for eliminating instrument noise in the GE plants and the fact that relay isolation devices are involved in these circuits in the GE plants, we concluded that one could reasonably expect similar acceptable test results from GE plants.

Rod Block Monitor

Our review of the rod block monitor revealed several concerns which were expressed in our Safety Evaluation Report.

We were concerned about the open spaces around the cables which penetrate the fire barriers betwen redundant channels of the four flow monitors. These open spaces have since been filled with steel filler plates to reduce the size of the openings to that of the flexible conduit. This is an acceptable resolution of this concern.

We also noted that the wiring of the rod block monitor bypass provided a potential flammable path between the A and B bypass circuits. This wiring has since been re-routed to provide acceptable separation.

We were concerned about the electrical isolation between the redundant rod block monitor channels and have since reviewed the tests of the devices which isolate the separate rod block monitor channels from the reactor manual control system, and hence, from each other. We conclude that these devices provide adequate isolation.

The rod block monitor subsystem contains multiplexing circuitry which interfaces with the reactor manual control system. The purpose of the rod block monitor subsystem is to supply an inhibit signal to the reactor manual control system to prevent control rod withdrawal if it is determined that a given rod withdrawal will exceed the minimum critical power ratio.

The multiplexing circuitry employed in the Zimmer rod block monitor and reactor manual control system, processes and transmits information about reactor status, control rod position, rod block logic, and rod control logic through common electrical signals. In earlier BWR designs this was accomplished by individual circuits. The new design has a self-testing capability to assure that this information is being processed correctly. We believe that the new multiplexing design is acceptable provided this self-testing capability is formally implemented through Technical Specifications. The Technical Specifications will be reviewed to confirm that this is done.

On the basis discussed above, we have concluded that the Zimmer Rod Block Monitor is acceptable.

Use of Nonsafety-Grade Equipment to Mitigate the Effects of Anticipated Transients

Systems not meeting all the criteria for safety-grade systems have been identified by the applicant to be involved in mitigation of certain anticipated transients. Of the systems so identified, only the following nonsafety-grade systems contribute to the mitigation of those potentially limiting transients which are analyzed to establish operating limits:

(1) The reactor manual control system mitigates the control rod withdrawal error at power.

(2) The feedwater control system high water level ("Level 8") trip mitigates the feedwater controller failure event.

(3) The turbine bypass system mitigates the feedwater controller failure event.

We have concluded that credit may be given for use of these systems in mitigation of the corresponding potentially limiting transients provided their operability is demonstrat periodically through surveillance requirements adopted into the Technical cifications.

Specifically, the reactor manual control system (Item 1) which performs an essential function associated with the rod block monitor in terminating control rod withdrawals at power, has a self-testing capability (See also Section 7.6.3).

We have required that the operability of this self-testing feature be demonstrated periodically.

The feedwater control system (Item 2) employs a high water level trip ("Level 8 trip") which terminates feedwater flow increases related to feedwater controller failure. We have concluded that there is adequate electrical isolation between the three level sensors (including their trip circuits) to assure that a single failure will not cause an excess feedwater transient and prevent the Level 8 trip from functioning. In addition, we required that the applicant adopt technical specifications to assure operability of the "Level 8 Trip."

The turbine bypass system (Item 3) is also involved in mitigation of feedwater controller failures. Although this system is considered to be reliable based on operating experience, it is not safety grade. We have concluded that credit may be taken for action of the turbine bypass system in analyzing the feedwater controller failure event based on adoption of technical specification surveillance requirements to confirm operability of the system.

Main Steam Isolation Valve Leakage Control System

The purpose of the main steam isolation valve leakage control system is to control and minimize the release of fission products, which would leak through the closed main steam isolation valves after a loss-of-coolant accident. This is accomplished by directing the leakage through a bleed line into an area served by the reactor building standby ventilation system for processing prior to release to the atmosphere.

The design of the leakage control system for the main steam isolation valves provides for two independent subsystems (upstream and downstream from outboard main steam isolation valves) powered from separate electrical divisions. The upstream and downstream subsystems each connect to the main steam lines through two isolation valves in series.

We noted that the serial isolation valves in either subsystem were powered from the same electrical division and thus, vulnerable to single failures which can result in the opening of both isolation valves during normal plant operation when the main steam isolation valves are open. Our review revealed that a single failure of a relay will cause the opening of both serial isolation valves. This problem has been corrected by modifying the design to include additional relays.

7.6.4 Evaluation

Based on the acceptable resolution discussed above (Section 7.6.3) of the concerns raised in Section 7.6 of our original Safety Evaluation Report, we consider the design of additional instrumentation systems required for safety at Zimmer to be acceptable.

7.7 Control Systems Not Required For Safety

7.7.3 Specific Findings

Control System Failures

With regard to the effects of control system failures or malfunctions, the analyses reported in Chapter 15 of the FSAR are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents, including those related to control systems. To provide assurance that the Chapter 15 analyses adequately bound events initiated by a single credible failure or malfunction, we have asked the applicant to provide additional information.

We have requested that a review be conducted to identify any power sources or sensors which provide power or signals to two or more control systems, and to demonstrate that failures or malfunctions of these power sources or sensors will not result in consequences outside the bounds of the Chapter 15 analyses or beyond the capability of operators or safety systems.

We have also requested a review by the applicant to determine whether the harsh environments associated with high energy line breaks might cause control system malfunctions and result in consequences more severe than those of Chapter 15 analyses or beyond the capability of operators or safety systems.

We will report on the resolution of these issues in a supplement to this report.

Reactor Manual Control System

We have identified the reactor manual control system as a system which is used to mitigate the effects of an anticipated transient but which does not meet all the criteria for safety grade systems. Our discussion of this issue is presented in Section 7.6.3 of this report. A discussion of the role of the reactor manual control system in providing rod position indication is presented in Section 7.5.3.

Feedwater Control System

We have identified the high water level trip ("Level 8 Trip") of the feedwater control system as a system which mitigates the effects of an anticipated transient but which does not meet all the criteria of a safety grade system. Our discussion of this issue is presented in Section 7.6.3 of this report.

7.7.4 Evaluation

With the exception of our concerns about control system failures which are discussed in Section 7.7.3 above, we consider the design of Zimmer control systems to be acceptable.

8 ELECTRIC POWER

8.1 Introduction

8.1.1 General Discussion

As a result of meetings with La Salle, the lead plant being reviewed for the utility owners of near-term boiling water reactors, it was brought to our attention that staff positions addressing degraded grid voltage and protection of reactor containment electrical penetrations were not transmitted to the applicant during the NRC staff review of the Zimmer FSAR. To correct this oversight, the positions were transmitted to the applicant by letter, dated September 12, 1980.

8.1.2 General Findings

Degraded Grid Voltage

The Millstone Unit 2 (Docket No. 50-336) low grid voltage occurrence brought into focus the potential common mode failure of redundant safety-related electrical equipment that could result from a degraded voltage condition. This occurrence prompted us to develop various positions to assure that the requirements of Criterion 17 of the General Design Criteria will not be violated.

The positions that we have developed are being used in the evaluation of electrical power designs for operating plants, as well as construction permit and operating license applications. The applicant has been made aware of these positions which are summarized as follows:

(1) In addition to the undervoltage scheme provided to detect loss of offsite power at the safety buses, we require that a second level of voltage protection for the onsite power system be provided with a time delay and that this second level of voltage protection shall satisfy the following criteria:

   (a) The selection of voltage and time setpoints shall be determined from an analysis of the voltage requirements of the safety-related loads at all onsite system distribution levels;

   (b) The voltage protection shall include coincidence logic on a per bus basis to preclude spurious trips of the offsite power source;

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

      (i) The allowable time delay, including margin, shall not exceed the maximum time delay that is assumed in the Final Safety Analysis Report accident analyses;

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

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

(d) The voltage sensors shall automatically initiate the disconnection of offsite power sources whenever the voltage setpoint and time delay limits have been exceeded;

(e) The voltage sensors shall be designed to satisfy the applicable requirements of Institute of Electrical and Electronics Engineers Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations"; and

(f) The Technical Specifications shall include limiting conditions for operation, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values for the second-level voltage protection sensors and associated time delay devices.

(2) We require that the current system designs automatically prevent load shedding of the emergency buses once the onsite sources are supplying power to all sequenced loads on the emergency buses. The design shall also include the capability of the load shedding feature to be automatically reinstated if the onsite source supply breakers are tripped. The automatic bypass and reinstatement feature shall be verified during the periodic testing identified in position 3 below.

(3) We require a test to demonstrate the full functional operability and independence of the onsite power sources at least once per 18 months during (a) loss of offsite power, (b) loss of offsite power in conjunction with an accident signal; and (c) interruption and subsequent reconnection of onsite power sources to their respective buses. Proper operation shall be determined by:

(a) Verifying that on loss of offsite power the emergency buses have been de-energized and that the loads have been shed from the emergency buses in accordance with design requirements;

(b) Verifying that on loss of offsite power the diesel generators start on the autostart signal, the emergency buses are energized with permanently connected loads, the auto-connected shutdown loads are energized through the load sequencer, and the system operates for five minutes while the generators are loaded with the shutdown loads;

(c) Verifying that on an accident signal (without loss of offsite power) the diesel-generators start on the autostart signal and operate on standby for five minutes;

(d) Verifying that on loss of offsite power in conjunction with an accident signal the diesel-generators start on the autostart signal, the emergency buses are energized with permanently connected loads, the auto-connected emergency (accident) loads are energized through the load sequencer, and the system operates for five minutes while the generators are loaded with the emergency loads; and

(e) Verifying that on interruption of the onsite sources the loads are shed from the emergency buses in accordance with design requirements and that subsequent loading of the onsite sources is through the load sequencer.

(4) We require that the voltage levels at the safety-related buses shall be optimized for the full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the intervening transformers. We also require that the adequacy of the design in this regard be verified by actual measurement and by correlation of measured values with analysis results. Before initial reactor power operation, documentation should be submitted verifying the adequacy of the design.

The applicant responded by letters dated October 31, 1980 and April 30, 1981 to the above positions. In regard to position 1, a second level of undervoltage protection is provided, consisting of two undervoltage relays and a timer (connected in a two-out-of-two coincident logic arrangement) at the 4160 V level on a per bus basis. When the setpoints of the undervoltage relays and timer have been exceeded, the power source to the essential safety features bus will automatically transfer from the offsite to the onsite source. This protection meets the position 1 requirement that a second level of undervoltage protection be provided, meets parts b, c, d and e of position 1 and is acceptable, pending review of final design implementation drawings.

Regarding part a of position 1, the voltage setpoint of 3220 V at 4160 V level has been selected, whereas Class 1E motors are rated for continuous operation between 3600 V to 4400 V. The applicant has committed, prior to fuel loading, to determine a new voltage setpoint based on a voltage analysis, and will demonstrate that no damage to Class 1E equipment will occur due to sustained undervoltage at this to-be-determined voltage level. We consider this resolved pending review of the documentation of the results of the analysis.

Regarding position 2, the applicant has documented that the existing load shedding system is adequately designed to prevent inadvertent load shedding. After the onsite sources are supplying power to sequenced loads (in response to an ECCS actuation signal) a load shed will occur only if:

(1) The voltage at the emergency bus dips below approximately 50% of the rated value.

(2) The voltage remains at this level for a predetermined time.

We conclude that the applicant's design is an acceptable alternative to that of our position.

Regarding position 3, the applicant has committed that the Technical Specifications will include a requirement to demonstrate the full functional operability and independence of the onsite power sources at least once during refueling shutdowns. The applicant further adds that the testing requirements will comply with the criteria defined by position 3. We find this acceptable and will verify implementation of the above commitments during our review of the Zimmer Technical Specifications.

In regard to position 4, the applicant has committed to perform a voltage analysis and verification test prior to fuel loading. We consider this acceptable pending review of the results of the analysis and test.

Protection of Reactor Containment Electrical Penetrations

Criterion 50 of the General Design Criteria requires, in part, that the reactor containment structures, including electrical penetrations, be designed so that the containment structure and its internal compartments can accommodate, without failure, the pressure and temperature conditions resulting from any loss-of-coolant accident. Therefore, with regard to electrical penetrations the main objective of our review is to determine that the electrical penetration assemblies are designed to withstand, without the loss of mechanical integrity, the maximum available fault current versus time conditions that could occur given single random failures of circuit overload protective devices as recommended by Regulatory Guide 1.63, Revision 1, "Electrical Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants."

In response to our concerns, the applicant documented by letter dated April 30, 1981 that:

"The penetration design will conform to position c.1 of Reg. Guide 1.63, Rev. 1 and with respect to backup overcurrent protection either;

"1. The overcurrent protection systems will conform to IEEE 279 (71); ANSI-N42.7 (72) or;

"2. Overcurrent devices upstream from the overcurrent device protecting the penetration will provide fully coordinated backup protection to maintain penetration integrity over the expected range of fault currents.

"Notes:

"1. The above response applies to power circuits only. Control and instrumentation circuits are not subject to detrimental high level fault currents.

"Implementation

"At the first refueling outage, a second means of overcurrent protection will be installed on those circuits where containment penetration integrity cannot be assured by overcurrent protection upstream from the overcurrent device currently protecting the penetration."

We consider the applicant's approach to protect reactor containment electrical penetrations for power circuits satisfactory. For control and instrumentation circuits we require the applicant to document maximum fault current versus time profiles and time current characteristics (I^2Rt) of the penetration conductors, to assure that sustained fault current levels will not cause any damage. Design implementation drawings should be submitted to verify that control power for primary and backup protective devices would be utilized from independent DC sources. We will condition the operating license upon the satisfactory resolution of this issue, prior to restart following the first refueling outage.

Station Blackout Events

A recent decision by the Atomic Safety and Licensing Appeal Board (ALAB-603) concluded that station blackout (i.e., loss of all offsite and onsite AC power) should be considered a design basis event for St. Lucie Unit No. 2. An amendment to the Construction Permit for St. Lucie Unit No. 2 was subsequently issued on September 18, 1980. The NRC staff is currently assessing station blackout events on a generic basis (Unresolved Safety Issue A-44). The results of this study, which is scheduled to be completed in 1982, will identify the extent to which design provisions should be included to reduce the potential for or consequences of a station blackout event.

However, the Board has recommended that more immediate measures be taken to ensure that station blackout events can be accommodated while task A-44 is being conducted. Although we believe that, qualitatively, there appears to be sufficient time available following a station blackout event to restore AC power, we are not sure if licensees have adequately prepared their operators to act during a station blackout event.

Consequently, we requested that the applicant review current plant operations to determine the capability to mitigate a station blackout event and promptly implement, as necessary, emergency procedures and a training program for station blackout events. The applicant's review of procedures and training should consider, but not be limited to:

1. The actions necessary and equipment available to maintain the reactor coolant inventory and heat removal with only DC power available, including consideration of the unavailability of auxiliary systems such as ventilation and component cooling.

2. The estimated time available to restore AC power and its basis.

3. The actions for restoring offsite AC power in the event of a loss of the grid.

4. The actions for restoring offsite AC power when its loss is due to postulated onsite equipment failures.

5. The actions necessary to restore emergency onsite AC power. The actions required to restart diesel generators should include consideration of loading sequence and the unavailability of AC power.

6. Consideration of the availability of emergency lighting, and any actions required to provide such lighting, in equipment areas where operator or maintenance actions may be necessary.

7. Precautions to prevent equipment damage during the return to normal operating conditions following restoration of AC power. For example, the limitations and operating sequence requirements which must be followed to restart the reactor coolant pumps following an extended loss of seal injection water should be considered in the recovery procedures.

The annual requalification training program should consider the emergency procedures and include simulator exercises involving the postulated loss of all AC power with decay heat removal being accomplished by natural circulation and the steam-driven auxiliary feedwater system for PWR plants, and by the steam-driven RCIC and/or HPCI and the safety-relief valves in BWR plants.

We conclude that the actions described above should be completed as soon as they reasonably can be (i.e., within 6 months). In addition, so that we may determine whether a license should be amended to incorporate this requirement, the applicant was requested, pursuant to § 50.54(f), to furnish an assessment of existing or planned facility procedures and training programs with respect to the matters described above. In the event that completion prior to licensing cannot be met, this requirement will be made a condition of the operating license.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.4 Fuel Handling Systems

Control of Heavy Loads

In January 1978, the NRC published NUREG-0410 entitled, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants - Report to Congress." As part of this program, the Task Action Plan for Unresolved Safety Issue Task No. A-36, "Control of Heavy Loads Near Spent Fuel," was issued.

We have completed our review of load handling operations at nuclear power plants. A report describing the results of this review has been issued as NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants - Resolution of TAP A-36." This report contains several recommendations to be implemented by all licensees and applicants to ensure the safer handling of heavy loads.

The applicant was requested by letter dated December 22, 1980 to review the controls for the handling of heavy loads to determine the extent to which the guidelines are presently satisfied at the Zimmer facility, and to identify the changes and modifications that would be required in order to fully satisfy these guidelines. Implementation of interim actions were also required by the December 22, 1980 letter. Resolution of this issue will be made a condition of the operating license.

9.2 Water Systems

9.2.1 Service Water System

Since the issuance of NUREG-0528, the applicant has experienced silting of the Ohio River intake structure which supplies service water for the Zimmer station. The staff reviewed this matter and the applicant's corrective action for any potential unreviewed safety issue. The staff's hydrology conclusions are discussed in subsection 2.4.3 of this supplement.

The applicant has provided ultrasonic transducers which alarm in the control room to continuously monitor silt buildup in the intake structure. In addition, a system of spray nozzles will be added to help keep silt in suspension and prevent deposition. Finally, a system for silt removal to the plants settling basin, periodically if required, will be employed. These systems are described in Appendix J to the Final Safety Analysis Report.

Technical Specifications will be established to ensure that continued operation with excessive silt will not occur.

The staff concludes that the measures taken to ensure that silt buildup will not impair critical water demands are acceptable.

9.5 Fire Protection Systems

In NUREG-0528, we stated that we would report the results of our evaluation of the Zimmer Nuclear Power Station fire protection systems. Our evaluation to date is contained in Appendix E of this supplement.

The review of the fire protection program for the Zimmer Nuclear Plant is not complete. The applican. has not yet committed to meeting the technical requirements of Appendix R to 10 CFR Part 50, or providing equivalent protection. We will resolve this issue in a subsequent supplement to this safety evaluation report.

9.6 Diesel Generator Systems

9.6.1 Diesel Generator Fuel Storage and Transfer System

By letter, dated December 27, 1979, the staff requested additional information on the Zimmer diesel generators (Q-020.39-52). The applicant responded to this request in Revision 64, February 1980, to the Final Safety Analysis Report. Our evaluation follows.

NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," made specific recommendations on increasing the reliability of nuclear power plant emergency diesel generators. Information requests concerning these recommendations, and also concerning the design of the fuel oil storage and transfer system, were transmitted to the applicant. The applicant responded in Revision 64 dated February 1980, stating how they meet or will meet the recommendations of NUREG/CR-0660 and our additional concerns.

We have reviewed these responses and have determined that conformance to the recommendations is as follows:

Recommendation	Conformance
1. Moisture in Air Start System	No
2. Dust and Dirt in D/G Room	Partial
3. Turbocharger Gear Drive Problem	No
4. Personnel Training	Yes
5. Automatic Prelube	No
6. Testing, Test Loading and Preventative Maintenance	Partial
7. Improve Identification of Root Cause of Failures	Yes
8. D/G Ventilation and Combustion Air Systems	Yes
9. Fuel Storage and Handling	Yes
10. High Temperature Insulation for Generator	*

*Explicit conformance is considered unnecessary by the staff in view of the equivalent reliability provided by the design, margin and qualification testing requirements that are normally applied to emergency standby diesel generators.

11.	Engine Cooling Water Temperature Control	Yes
12.	Concrete Dust Control	Partial
13.	Vibration of Instruments and Controls	Yes

On the basis of our review we have concluded that there is sufficient assurance of diesel generator reliability to warrant unrestricted plant operation through the first refueling period. However to assure long term reliability of the diesel generator installations we require that the following design and procedural modifications be implemented prior to startup following the first refueling.

1. Moisture in Air Starting System: The Air Starting System at Zimmer does not include air dryers to remove moisture. The system relies on manual blow down valves on the receivers to reduce the moisture in the system. Operating experience has shown that accumulation of water and other contaminants in the starting system have been the most frequent causes of diesel engine failure to start on demand. In an effort toward improving their starting reliability we require that the air be dried to a dew point of not more than 50°F when installed in a normally controlled environment, otherwise the starting air dew point should be controlled to at least 10°F less than the lowest expected ambient temperature. Modify the present design of the air starting system to include an air dryer to provide clean and dry air to the diesel engine air start valves.

2. Dust and Dirt Control in the D/G Room: The applicant has stated that auxiliary relays, control switches, etc., are located in drip-proof free-standing control panels. All control devices, i.e., relays, switches, have their contact mechanism either enclosed within the device frame or enclosed within a separate device enclosure. Only the terminals are exposed. The applicant also stated that the panels were not designed to be dust proof. However, he has committed to investigate the panel construction to see if with minor modifications (i.e., addition of gaskets, rubber seals) they can be improved so that adequate protection of the control devices from dust accumulation can be achieved. In the event the proposed minor field modification to the panels do not result in adequate protection of control devices from dust accumulation, additional protection will be provided as necessary at the control device to achieve an adequate dust seal. We find this acceptable.

3. Turbocharger Gear Drive Problem: The diesel generators at Zimmer have a Turbocharger Mechanical Drive Gear Assembly whose gear ratio is 18:1. This drive gear assembly has not been designed to operate at no load or light load conditions and full rated speed for prolonged periods. The manufacturer, Electro-Motive Division of General Motors Corporation (EMD) has developed or has under development heavy duty turbocharger drive gear assemblies which meet the recommendation of NUREG/CR-0660. To improve the reliability and availability of the diesel generators on demand we require the installation of a heavy duty turbocharger drive gear assembly as recommended by NUREG-0660.

4. Automatic Prelube: The lubrication system for the Zimmer diesel engine is composed of a continuously operating alternating current pump and a standby direct current pump which prelubricate the turbocharger bearings only. The other wearing parts of the engine do not receive any lubrication until after the engine starts, and the engine driven lube oil pump reaches

full speed. This is not acceptable. We require a prelube pump since dry starting of the diesel-generators under emergency conditions will result in momentary lack of lubrication at various moving parts which can eventually lead to failures with resultant equipment unavailability. We require that this pump shall be used for all modes of diesel engine starting and be capable of providing lubrication to all wearing parts of the engine. The objective is to improve the availability of this equipment on demand. The pump shall be powered from a reliable direct current power supply and installed in the system to operate in parallel with the engine driven lube oil pump. In an automatic start, the prelube pump should operate only during the engine cranking cycle or until a satisfactory lube oil pressure is established in the engine main lube oil distribution header. The prelube pump should also be provided with manual start.

5. No Load Operation: The applicant stated that the diesel generator sets at Zimmer may be run continuously unloaded if a reactive load is applied to the generator (until stabilized) every 2 to 3 hours, but he did not provide any no load operating procedures. We find this unacceptable. We require that operating procedures be developed that require loading the diesel engine to a minimum of 25 percent of full load for one hour after 2 to 3 hours of continuous no load operation or as recommended by the engine manufacturer. The present diesel generator design meets the requirements of General Design criteria 17, 18 and 21 of Appendix A of 10 CFR Part 50. Upon completion of the above changes and modifications, the design of the diesel generator and its auxiliary systems will also be in conformance with recommendations of NUREG/CR-660 for enhacement of diesel generator reliability and the related NRC guidelines and criteria. We therefore conclude that this will provide assurance of diesel generator reliability through the design life of the plant. Full implementation of our position on this matter will be made a condition of the operating license.

9.6.2 Diesel Generator Auxiliary Systems

(See subsection 9.6.1 above)

10 STEAM AND POWER CONVERSION SYSTEM

10.2 Turbine Generator

Turbine Disc Integrity

We have reviewed Section 10.2.3 of the Final Safety Analysis Report for the Wm. H. Zimmer Nuclear Power Station No. 1 and conclude that the integrity of the turbine will be adequate and that reasonable assurance is provided that the applicable parts of General Design Criteria of 10 CFR Part 50 will be met.

The turbine discs and rotors are forged from vacuum degassed steel by processes that minimize flaws and provide adequate fracture toughness. These materials have the lowest fracture appearance transition temperatures and highest Charpy V-notch energies obtainable on a consistent basis. The maximum tangential stress in discs and rotors resulting from centrifugal forces, interference fit and thermal gradients does not exceed 0.75 of the yield strength of the materials at 115% of the rated speed.

The preservice inspection program calls for 100% ultransonic inspection (UT) of each rotor and disc forging before finish machining and magnetic particle (MT) after finish machining. No MT flaw indications are permissible in bores, holes, keyways and other highly stressed regions.

Since 1979 the staff has known of the stress corrosion problems in low pressure rotor discs in Westinghouse turbines. Appropriate conservative inspection intervals have been effective in monitoring crack growth to permit repair or replacement of discs well in advance of failure. The applicant has submitted to the staff the material properties of the low pressure turbine discs, as well as the calculations of critical crack sizes. The method used to predict crack growth rates is based on evaluating all of the cracks found to date in Westinghouse turbines, past history of similar turbine disc cracking, and results of laboratory tests. This prediction method takes into account two main parameters; the yield strength of the disc, and the temperature of the disc at the bore area where the cracks of concern are occurring. The higher the yield strength of the material and the higher the temperature, the faster the crack growth rate will be.

We have evaluated the data submitted by the applicant, and in addition, performed our own calculations for crack growth and critical crack size. We conclude that Zimmer Unit 1 may be safely operated until the first refueling outage, at which time the LP turbine discs should be inspected.

Inservice inspection will include UT of the bore and keyway areas of each disc and MT and visual inspection of all accessible areas.

The turbine meets our criteria regarding the use of materials with acceptable fracture toughness and adequate design. Preservice and inservice inspection criteria are in accordance with current staff guidelines. The materials, processes and designs used by the applicant are therefore considered acceptable. We conclude that these provisions provide reasonable assurance that the probability of disc failure with missile generation is low during normal operation, including transients up to design overspeed.

12 RADIATION PROTECTION

12.1 Shielding

Fuel Transfer Process

By letter, dated May 18, 1979, the staff requested additional information regarding biological shielding in the vicinity of the spent fuel transfer process (Q-331.21). The applicant responded in Revision 59, July 1979, to the Final Safety Analysis Report but said that additional studies of the matter were in progress. The applicant provided additional information in FSAR Amendment 112, Revision 69.

The refueling floor area is protected from high radiation levels during refueling by water and permanent concrete shielding. In order to assure acceptable radiation levels in the upper drywell area during refueling, Zimmer will use a portable fuel transfer shield. This shield will span the gap between the reactor pressure vessel and the containment wall and will be used during the transfer of all fuel assemblies which have experienced some burnup. Use of this shield will reduce the peak dose rate from normal transfer of spent fuel to less than 100 mr/hr at an elevation 9'6" above the highest drywell grating (15 mrem per assembly when fuel is traveling at its slowest transfer speed).

In addition to this portable shield, personnel will be prohibited from entering the upper drywell area during initial fuel transfer. Portable, alarming area radiation monitoring equipment will be placed in the upper drywell area to measure dose rates during spent fuel transfer. We find that Zimmer has taken adequate steps to prevent inadverted exposures during spent fuel transfer.

12.3 Health Physics Program

The Health Physics Program objective is to provide administrative control of onsite personnel to assure that occupational radiation exposures are within the limits of 10 CFR Part 20 and are as low as is reasonably achievable, consistent with the intent of Regulatory Guide 8.8 and other applicable regulatory guides. The Rad/Chem Supervisor has the responsibility for administering this Program. He influences decisions on plant operation that can affect the radiation safety of workers, in that he is a member of the Station Review Board and he reports directly to the Assistant Plant Superintendent. The Rad/Chem Supervisor at Zimmer meets the minimum requirements of Regulatory Guide 1.8 (September 1975), "Personnel Selection and Training," which references ANSI 18.1 (1971) for a plant RPM. The senior rad/chem technicians at Zimmer also meet the minimum experience requirements of ANSI 18.1.

The Health Physics Program at Zimmer is designed to ensure that: (1) operations, maintenance, and technical personnel are trained to the extent required for their duties, consistent with 10 CFR Parts 19 and 20 and Regulatory Guide 1.8; (2) detailed procedures are prepared and approved for all aspects of the Radiation Protection Program; (3) appropriate access control procedures are followed to separate potentially contaminated areas from clean areas, and that positive control is provided for each entry in a high radiation area; (4) potential transfer of radioactive contamination is controlled by monitoring personnel, equipment, tools, and clothing; (5) radiation levels are measured and posted and personnel are monitored and provided with appropriate bioassay; (6) complete radiation exposure records are maintained; and (7) personnel access to high radiation areas and maintenance work in radiation areas are controlled by use of a Radiation Work Permit, which must be approved by the Rad-Chem Supervisor and which specifies any special requirements for the job.

Zimmer has a comprehensive training and qualification program for their rad/chem technicians. The object of this program is to train the technicians in the fields of health physics, chemistry, and radwaste and maintain their levels of proficiency in these two areas.

Most of the training subject areas addressed in Appendix E of draft NUREG-0761, "Radiation Protection Plans for Nuclear Power Reactor Licensees," are included in Zimmer's training schedule. We conclude that Zimmer's training program, as described in their procedures, provides an acceptable method for rad/chem technicians at Zimmer to be able to maintain an adequate understanding of and perform both chemistry and radiation protection functions at Zimmer.

The rad/chem technicians hired at Zimmer all have between 1 and 4 years of college background. Because of the high caliber of people hired and because of the comprehensive training program at Zimmer, Zimmer has had a high retention rate of rad/chem technicians. Rad/chem technicians at Zimmeer will be trained to perform chemistry, health physics, and radwaste functions. Typically, senior technicians will be assigned to a rotating schedule in which they will spend several weeks at a time performing work in each of these fields. Technicians and junior technicians will rotate among the three disciplines on a more frequent basis. This continuous rotation of functions should allow the technicians to maintain their expertise in all three areas. The liquid and solid radwaste systems at Zimmer will be operated by senior rad/chem technicians, qualified in their use, instead of by entry level nuclear plant operators as part of their training program, as is often done at other plants. By using people familiar with the use and function of the radwaste system, Zimmer hopes to minimize releases and properly process solid radioactive waste.

The radiation protection facilities at Zimmer include access control points, high and low level laboratories, counting room, instrument calibration room, offices, decontamination and laundry area, and change room. Based on our review, we conclude that these facilities are sufficient to maintain occupational exposures as low as is reasonably achievable, and are consistent with Regulatory Guide 8.8.

The applicant will provide equipment to be used for radiation protection which includes: protective clothing, respiratory protective equipment, air sampling equipment, portable radiation measuring instruments, calibration sources, counting room instrumentaton, area monitors, airborne activity monitors, laboratory equipment and special shielding materials. Based on our review, we conclude that the numbers and types of this equipment will be adequate to provide reasonable assurance that exposures to personnel can be maintained as low as is reasonably achievable.

All persons entering a restricted area are provided with a TLD to monitor beta-gamma radiation. Persons entering a radiation area are also provided with a self-reading dosimeter. A neutron badge is provided for personnel who enter areas where the rate exceeds 5 millrems per hour or could exceed 15 millirems during a given month. Finger rings, wrist badges, or other dosimeters, as well as alarming dosimeters, are provided as appropriate. Whole-body counting will be performed periodically to assess intake of radioactive materials. Bioassay may also be used as necessary.

Based on the information provided in the application and the responses to our questions, we conclude that the applicant intends to implement a radiation protection program that is acceptable and will keep radiation exposures as low as is reasonably achievable.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure and Qualifications

The applicant has recently reorganized its corporate management structure to place all of the operations and most of the engineering support activities related to the Zimmer Station under a single Vice President, Nuclear Operations who reports to E. A. Borgmann, the Senior Vice President, Engineering Services and Electric Production. Reporting to the Vice President, Nuclear Operations in this new management structure are four corporate managers: the Manager, Wm. H. Zimmer Station; the Manager, Nuclear Engineering; the Manager, Quality Assurance; and the Manager, Nuclear Services. The applicant has documented these organizational changes in Revision 72 to the Zimmer FSAR.

Prior to the reorganization as discussed above, the corporate engineering and quality assurance support for the Zimmer Station was provided by corporate function organizations that were not dedicated solely to the Zimmer Station. They also provided support service to the applicant's fossil fuel plants. The New Nuclear Operations organization will be dedicated solely to the support of nuclear station activities.

Technical support for licensing, environmental and emergency planning activities for the Zimmer Station are provided and managed by the corporate Licensing and Environmental Affairs organization. Construction activities related to the Zimmer Station are managed by the corporate General Construction organization. Both Licensing and Environmental Affairs and General Construction provide services to fossil stations as well as to the Zimmer Station, and they both report directly to the Senior Vice-President, Engineering Services and Electric Production.

Under the new organization, the Zimmer Station Superintendent has been redesignated as the Manager, Wm. H. Zimmer Station. The organization at the site reporting to this Manager remains essentially as it was before except that the onsite quality assurance staff and the training coordinator who previously reported to the Station Superintendent will now report offsite to the new Manager, Quality Assurance and the new Manager, Nuclear Services, respectively.

The size of the organization at the site (the station staff) has been substantially increased over that previously reported in the Zimmer SER. Revision 72 to the Zimmer FSAR indicates the typical staffing levels are: operations, 36 to 48; radiation protection and chemistry, 20 to 24; instrument and controls, 14 to 18; maintenance 37 to 42; engineering, 10 to 14; other personnel, 17 to 26. The minimum shift crew size has increased to six operators, reflecting the addition on each crew of another licensed senior reactor operator as required by TMI Action Plan Item I.A.1.3. A shift technical advisor will be onsite at all times when the nuclear plant is operating in Modes 1, 2 or 3 as required by TMI Action Plan Item I.A.1.1. As before, a radiation chemist will be onsite at all times.

The Zimmer SER stated that a Maintenance Supervisor had not, at that time, been selected and that the NRC staff would review the qualifications of this individual whenever the position was filled. The position was filled in mid-1979. We have reviewed the qualifications of this Maintenance Supervisor and concluded that they meet the requirements of Regulatory Guide 1.8, Revision 1, and are acceptable.

The new Nuclear Engineering organization will be located at the corporate office and will provide support for the Zimmer Station in most engineering disciplines.

The new Quality Assurance organization will be located at the corporate office and will be responsible for all Zimmer related quality control and quality assurance. As discussed above, the quality assurance staff that previously reported to the Station Superintendent will now report directly to the Manager, Quality Assurance.

The new Nuclear Services organization will be located at the corporate office and will provide services for the Zimmer Station in the areas of reliability analysis, nuclear fuel, nuclear systems, and training. It will also be responsible for the activities of the Independent Safety Review Group (ISRG) which the applicant has established in response to TMI Action Plan Item I.B.1.2. This ISRG is located at the Zimmer Station but reports offsite to the Manager, Nuclear Services.

Revision 72 to the Zimmer FSAR indicates that typical engineering staffing levels for the three new organizations are: Nuclear Engineering 15 to 17; Quality Assurance, 10 to 13; and Nuclear Services, 18 to 25.

We conclude that the revised and augmented organization for operation and support of the Zimmer Station as further discussed in Action Plan Item I.B.1.2 should increase the capability of the applicant to assure safe operation of the Zimmer Station and is acceptable.

13.2 Training Program

The information contained in NUREG-0528, SER for the Zimmer Nuclear Power Station Unit 1, remains valid. The training program provides a flexible, effective means of preparing personnel for the preoperational test program, for operator licensing examinations, and for fuel loading.

Standard Review Plan 13.2, "Training," was used in the review. Regulatory Guide 1.8, "Selection and Training of Personnel," and 10 CFR 55.20 through 55.23 and 55.25 were used to evaluate the programs.

13.3 Emergency Planning

Emergency Planning requirements have been significantly changed as a result of TMI-2. They are discussed in Section 22 of this supplement.

13.4 Review and Audit

Revision 72 to the Zimmer FSAR indicates that the new Vice President, Nuclear Operations and each of the four managers reporting to him (Manager, Zimmer Station; Manager, Nuclear Engineering; Manager, Quality Assurance; and Manager, Nuclear

Services) will serve as a member of the Offsite Review Committee (ORC), the applicant's independent review body. In addition, three additional corporate office managers (Manager, General Engineering; Manager, Licensing and Environmental Affairs; and Manager, Fossil Production) and two outside consultants also serve as members of the ORC. The Senior Vice President, Engineering Services and Electric Production is the chairman of the ORC.

The two outside consultants were added to the ORC membership in response to the March 13, 1979 ACRS recommendation that the ORC "include additional experienced personnel from outside the corporate structure as voting members for the first few years of operation."

Based on our review of the subject, we conclude that the applicant has complied with this ACRS recommendation and that the revised membership of the ORC as proposed by the applicant is acceptable.

13.7 Industrial Security

The applicant has submitted security plans entitled "Wm. H. Zimmer Nuclear Power Station Industrial Security Plan" and "Wm. H. Zimmer Nuclear Power Station Guard Training and Qualification Plan," for protection against radiological sabotage. The contingency plan is incorporated as Chapter 8 in the Wm. H. Zimmer Nuclear Station Industrial Security Plan. The staff has reviewed these documents and has identified certain areas that require changes in order to comply with the requirement of Section 73.55 of 10 CFR Part 73. The applicant has been informed of the areas of the security plan, contingency plan, and guard training and qualification plan requiring revision. It is concluded that the applicant's security plan, contingency plan, and guard training and qualification plan, when formally modified to incorporate the required changes will be acceptable.

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 based on a confirmatory evaluation of the plant to determine those areas where acts of sabotage might cause a release of radionuclides in sufficient quantities to result in dose rates equal to or exceeding 10 CFR Part 100 limits.

The applicant's security plan is being withheld from public disclosure in accordance with Section 2.790(d) of 10 CFR Part 2.

14 INITIAL TEST PROGRAMS

14.1 Initial Test and Operation

14.1.1 Preoperational Test Program

Test of Essential Direct Current Systems

In NUREG-0528 we stated, "The applicant's proposed tests of the essential direct current systems do not include demonstration of the capability of essential loads to operate at the direct current systems design bases minimum voltage level. The applicant submitted some justification for omitting this testing in Final Safety Analysis Report, Revision 46 (Amendment 76). We reviewed this information and concluded that the justification is not adequate. We will require either that the applicant include these demonstrations in his preoperational tests of the 125-volt and 250-volt direct current systems or provide further technical justification for their omission. We will report resolution of this matter in a supplement to this report."

In Revision 55 of the Final Safety Analysis Report, the applicant described his plan to demonstrate the operability of at least one of each type of load (e.g. relay, feeder breaker) on a 125-volt bus at the minimum design basis voltage. We conclude that this is an acceptable method of verifying the operability of the 125-volt direct current system loads at low voltage. The 250-volt direct current system does not supply power to any engineered safety feature loads. The system does supply power to a reactor core isolation cooling system motor-operated valve, but this valve is assumed to operate only in the early stages of the design basis accident sequence. Therefore, we conclude that testing of the 250-volt system loads at minimum design basis voltage levels is not required.

Full-load Tests of Transformers

By letter, dated January 17, 1980, the staff requested the applicant confirm that full-load tests of 4160-480 volt transformers supplying vital buses will be conducted in accordance with Regulatory Guide 1.68, "Initial Test Programs For Water-cooled Nuclear Power Plants."

The applicant's response to the staff position stated that the testing will be conducted in accordance with Position 3 of Regulatory Guide 1.68, November 1973, "Preoperational and Initial Startup Test Programs for Water-Cooled Reactors," and also in accordance with part 4 of the staff position on degraded grid voltage. Based on the commitment to conduct the test in accordance with Regulatory Guide 1.68, we conclude that the response is acceptable. The detailed test method and acceptance criteria will be reviewed as part of the degraded grid voltage issue. The resolution of that issue is addressed in Section 8.1 of this SER supplement.

14.1.2 The Startup Test Program

In NUREG-0528 we noted that because of changes to the test program recommended by the General Electric Company, the applicant had not submitted revised descriptions of several of the startup tests until recently. We have received the needed information and have issued our positions on this matter. We also have received a complete response to our positions. Our conclusion regarding the startup test program is that it will be conducted in accordance with Section 14.2 of the Standard Review Plan and is acceptable. Additional low power testing in accordance with Item I.G.1 of NUREG-0737 is discussed in Section 22 of this supplement.

15 ACCIDENT ANALYSIS

15.1 Abnormal Operational Transients

Analysis of Operational Transients

In NUREG-0528, it was explained that many abnormal operational transients were analyzed with the methods described in NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactors" (the REDY Code) which was then under review. In this regard, three turbine trip tests were performed at the Peach Bottom Unit 2 plant for the purpose of providing experimental data for code verification, and to improve the understanding of integral plant behavior under transient conditions. Results from this test program raised some questions about the analytical methods then in use since r all the test data were conservatively predicted by the current licensing methods. As a result, the General Electric Company developed a new computer code called ODYN to more adequately model overpressurization transients. The ODYN code has been reviewed by the staff and found acceptable. (Safety Evaluation for the General Electric Topical Report-Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDE-24154-P Volumes I, II, III, June 1980.)

In March 1979, the applicant was requested to reanalyze the following transients using ODYN:

(1) For Thermal Limit Evaluation

- (a) Feedwater controller failure - maximum demand,
- (b) Generator load rejection without bypass, and
- (c) Turbine trip without bypass.

(2) For American Society of Mechanical Engineers Overpressure Protection

- (a) Main steam isolation valve closure with position switch scram failure (main steam isolation valve closure with flux scram).

The applicant has performed the required ODYN reanalysis. The results of this reanalysis show that for thermal margin considerations, all events are bounded by the rod withdrawal error at power. Accordingly, this event will be used to establish the operating limit minimum critical power ratio thereby providing assurance that the safety limit will not be violated by any of the abnormal operating transients analyzed. The results of the overpressurization analysis for the American Society of Mechanical Engineers Code compliance showed that peak vessel pressure at the bottom of the vessel when all 13 of the safety valves are considered is 1270 pounds per square inch gauge, which is below the code limit of 1375 pounds per square inch gauge for Zimmer. In addition, the ODYN reanalysis of the turbine trip without bypass and generator load rejection without bypass events did not take credit for the level 8 trip. The applicant agreed to propose a modification to the technical specifications with regard to the availability, setpoints, and surveillance requirements for the level 8 trip and the turbine bypass system. In conclusion, we find the applicant's response and the results of the analyses of abnormal operational transients acceptable.

Failure of Feedwater Heater

The applicant's FSAR analysis for the failure of the feedwater heater indicated that the temperature decrease is no greater than 100°F. At a domestic boiling water reactor, an actual temperature transient occurred which demonstrated a temperature decrease of 150°F. We required that the applicant reanalyze this event for such a temperature change, or justify the temperature decrease used in a reanalysis if other than 150°F.

Because the drop in feedwater temperature results in a reactor power level scram, the differences in peak surface heat flux and reduced MCPR are negligible between a 100°F and 150°F feedwater temperature decrease. The only difference between the two transients, as shown from the reanalysis, is that the power level scram occurs marginally earlier for the 150°F temperature change due to the greater reactivity insertion. Results of the reanalysis have adequately demonstrated that the conclusions reached previously in the FSAR concerning feedwater heater failures remain applicable.

15.2 Accidents

Anticipated Transients Without Scram

Background

Anticipated transients without scram (ATWS) are events in which the scram system (reactor trip system) is postulated to fail to operate as required. This subject has been under generic review by the Commission staff for several years.

In December 1978, Volume 3 of NUREG-0460, "Anticipated Transient Without Scram for Light Water Reactors," was issued describing the proposed type of plant modifications we believe are necessary to produce the risk from anticipated transients with failure to scram to an acceptable level. We issued requests for the industry to supply generic analyses to confirm the anticipated transients without scram mitigation capability described in Volume 3 of NUREG-0460, and subsequently we presented our recommendations on plant modifications to the Commission in September 1980. The Commission will determine the required modifications to resolve anticipated transient without scram concerns as well as the required schedule for implementation of such modifications. Zimmer is subject to the Commission's decision in this matter.

It is our expectation that the necessary plant modifications will be implemented in one to four years following a Commission decision on anticipated transients without scram. As a prudent course, to further reduce the risk from anticipated transient without scram events during the interim period before completing the plant modifications determined by the Commission to be necessary, we require that the following steps be taken:

1. An emergency operating procedure should be developed for an anticipated transient without scram event, including consideration of scram indicators, rod position indicators, average power range flux monitors, reactor vessel level and pressure indicators, relief valve and isolation valve indicators, and containment temperature, pressure and radiation indicators. The emergency operating procedures should be sufficiently simple and unambiguous to permit prompt operator recognition of an anticipated transient without scram event.

2. The emergency operating procedure should describe actions to be taken in the event of an anticipated transient without scram including consideration of manually scramming the reactor by using the manual scram buttons, changing the operation mode switch to the shutdown position, tripping the feeder breakers on the reactor protection system power distribution buses, scramming individual control rods from the back of the control room panel, tripping breakers from plant auxiliary power source feeding the reactor protection system, and valving out and bleeding off instrument air to scram solenoid valves. These actions must be taken immediately after detection of an ATWS event. Actions should also include prompt initiation of the residual heat removal system in the suppression pool cooling mode to reduce the severity of the containment conditions; and actuation of the standby liquid control system if a scram cannot be made to occur.

Early operator action as described above, in conjunction with the recirculation pump trip which has already been approved and installed at Zimmer, would provide significant protection for some ATWS events, namely those which occur (1) as a result of common mode failure in the electrical portion of the scram system and some portions of the drive system, and (2) at low power levels where the existing standby liquid control system capability is sufficient to limit the pool temperature rise to an acceptable level.

We have reviewed the Zimmer ATWS procedure and have concluded that these procedural requirements have been incorporated. This provides an acceptable basis for licensing and interim operation of Zimmer pending the outcome of the Commission rulemaking on ATWS in accordance with General Design Criteria 10, 15, 26, 27, and 29 of 10 CFR 50 Appendix A. The Commission will, by rulemaking, determine any future modifications necessary to resolve ATWS concerns and the required schedule for implementation of such modifications.

17 QUALITY ASSURANCE

17.2 Quality Assurance Program

Zimmer Q-List Review

Our review of the quality assurance program description for the operations phase for the Zimmer Nuclear Power Station has verified that the criteria of Appendix B to 10 CFR Part 50 have been adequately addressed in Chapter 17 of the FSAR. This determination of acceptability included a review of the list of items to which the quality assurance program applies.

The list of items was reviewed by the technical review branches to assure that safety-related items within their scope of review fall under the quality assurance program controls. Differences between the staff and the applicant regarding the list have been resolved to the staff's satisfaction. The list has been expanded to include safety-related items reflected in NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980. Therefore, the staff has no open items concerning the quality assurance program for operations or to what the program applies.

18 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

In its report dated March 13, 1979 on the Wm H. Zimmer Nuclear Power Station Unit 1 (see Appendix B), the Advisory Committee on Reactor Safeguards noted the following topics to which it believes the applicant and staff should give due consideration.

The Committee believes that our Mark II containment acceptance criteria is suitable for the lead Mark II plants. Zimmer is the lead Mark II plant. However, since the applicant had taken some exception to our criteria, it was necessary for us to review these exceptions and their bases prior to complete resolution of this matter. The Committee wished to be kept informed on the outcome of this additional review (see subsection 6.2 of this supplement).

Because of recent test performed in the Two Loop Test Apparatus, the Committee believes that some modification of the present emergency core cooling system analytical model may be necessary. However, we have concluded that the present model has adequate margins for assessing the performance of the Zimmer plant. The Committee wished to be kept informed on this matter (see subsection 6.3.4 of this supplement).

In view of the important role of the Operations Review Committee, the Advisory Committee recommended that the Operations Review Committee include additional experienced personnel from outside the corporate structure as voting members for the first few years of operation. The applicant has adopted the recommendation and has provided us with the names and qualifications of additional members. The staff finds the applicant's Operations Review Committee acceptable (see Subsection 13.4 of this supplement).

Regarding the status of generic items relating to the light water reactors, listed in the Advisory Committee's report, we consider II B-4 and II C-5 to be resolved for Zimmer and II C-1 to be not applicable to Zimmer. The remaining items will be dealt with by the applicant and staff as solutions are delineated.

The Advisory Committee concluded that if due consideration is given to the items mentioned in its March 13, 1979 report and subject to satisfactory completion of construction and preoperational testing, the Wm. H. Zimmer Nuclear Power Station, Unit 1, can be operated without undue risk to the health and safety of the public.

20 FINANCIAL QUALIFICATIONS

20.1 Introduction

In the Safety Evaluation Report dated January 1979, we prepared an evaluation of the financial qualifications of Cincinnati Gas and Electric Company, Columbus and Southern Ohio Electric Company, and Dayton Power and Light Company which concluded that there was a reasonable assurance that the subject companies could obtain the necessary funds to cover the estimated costs of operating the Zimmer Nuclear Power Station, Unit No. 1 and, if necessary, permanently shut the facility down and maintain it in a safe condition. The purpose of this report is to supersede that evaluation by including 1980 historical data and performing the analysis on the basis that 1983 would be the first full year of commercial operation. This analysis is a summary of the staff's testimony addressing the applicants' financial qualifications to operate and if necessary permanently shut down the Zimmer Nuclear Power Station and maintain it in a safe condition as presented in the Zimmer operating License bearings.

The Nuclear Regulatory Commission's regulations relating to the determination of an applicant's financial qualifications for a facility operating license appear in Section 50.33(f) and Appendix C to 10 CFR Part 50. At our request, Cincinnati Gas and Electric Company, Columbus and Southern Ohio Electric Company, and Dayton Power and Light Company submitted updated financial information regarding estimated operating and decommissioning costs for the Zimmer Nuclear Power Station, Unit No. 1, along with additional material covering the applicants' financial status. The following analysis summarizes our review of this submittal and addresses each applicant's financial qualifications to operate and, if necessary, permanently shut down and safely maintain the subject facility.

20.2 Business of Applicants

Cincinnati Gas & Electric Company is a corporation organized and operating under the laws of the State of Ohio with its principal place of business located at Fourth and Main Streets, Cincinnati, Ohio 45202. It is an investor-owned public utility whose primary business is to meet the electric and gas service requirements of some 583,000 electric customers and 361,000 gas customers in southwestern Ohio.

Columbus & Southern Ohio Edison Company is a corporation organized and operating under the laws of the State of Ohio with its principal place of business located at 215 North Front Street, Columbus, Ohio 43215. Likewise, it also is an investor-owned public utility whose primary business is to meet the electric requirements of some 457,000 customers in central and southern Ohio.

Dayton Power & Light Company is a corporation organized and operating under the laws of the State of Ohio with its principal place of business located at 25 North Main Street, Dayton, Ohio 45401. It is also an investor-owned public utility whose primary business is to meet the electric and gas service requirements of some 417,000 electric and 263,000 gas customers in southwestern Ohio.

20.3 Estimated Operating Costs of Facility

For the purpose of estimating the facility's operating costs, the applicants have assumed that 1983 would will be the first full year of commercial operation. Estimates of the total annual cost of operating the Zimmer plant for each of the first five years are presented in Table 20-1 below. As an element of conservatism, we requested that the applicants provide additional estimates of operation expenses for the facility's first five years under separate capacity factors with the assumptions of 50 percent and 60 percent. These are likewise stated in Table 20-1 below. All operating estimates for Zimmer cost are based upon a peak net electrical capacity of 792 megawatts and total estimated construction costs of $1,067.3 million. As of October 15, 1980, the Zimmer facility was 94.2 percent complete in construction. Operating costs include all costs associated with the capital investment and operation and maintenance including nuclear fuel.

20.4 Estimated Shutdown Costs of Facility

The applicants based their estimate of decommissioning costs upon a report published in November 1978 by the Atomic Industrial Forum (AIF) entitled, "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives." This study provided cost estimates in 1975 dollars for several decommissioning alternatives for BWR plants of both 1160-MWe and 550-MWe sizes. The AIF study concludes that the most economical mode of decommissioning would be either temporary mothballing or temporary entombment for a cooling period of about 104 years, followed by dismantling and removal of the radioactive structures of the facility.

If it is assumed that a security force will be required to guard a temporarily mothballed facility for the entire 104-year cooling period, then temporary entombment becomes the more economical choice. For purposes of these cost estimates, it was assumed that such a security force would be required for temporary safe storage without physical entombment.

There are four components of total decommissioning cost under the applicants' proposals: (1) cost of prompt dismantling and removing nonradioactive structures at the end of the 33-year life; (2) initial entombment cost; (3) annual surveillance and maintenance costs for the next 104 years; and (4) cost of dismantling and removing remaining structures at the end of the 104-year cooling period. No further expenses are expected to be incurred after the final dismantlement and removal. The AIF estimates for these four components for an 1160-MWe BWR and for a 550-MWe BWR were interpolated by the applicants to obtain the estimates for the 800-MWe Zimmer plant. Credit for the value of the land after completion of the decommissioning project was also estimated. These applicant estimates are listed in Table 20-2 in 1983 dollars. A 6 percent annual inflation rate has been assumed from 1975 through 1979. A 6.5 percent annual inflation rate has been assumed after 1979.

At the request of the NRC, an indepth review of the AIR report "An Engineering Evaluation of Nuclear Power Reactor Decommissioning Alternatives" (AIF/NESP-009 and -009 SRs) was performed by Battelle Pacific Northwest Laboratories. In Battelle's November 18, 1977 review, particular attention was paid to the estimation of costs, their bases, and the methodology used. As to the overall

Table 20-1 Estimate of total annual cost of operation of William H. Zimmer Nuclear Generation

Year	1983	1984	1985	1986	1987
Applicants' Estimate					
Plant Capacity Factor (percent)	52.0	63.0	76.0	74.0	78.0
Annual Cost of Operation (millions)	\$252.5	\$273.1	\$260.7	\$252.8	\$260.4
Alternative I					
Plant Capacity Factor (percent)	50.0	50.0	50.0	50.0	50.0
Annual Cost of Operation (millions)	\$251.2	\$263.4	\$243.9	\$236.9	\$242.1
Alternative II					
Plant Capacity Factor (percent)	60.0	60.0	60.0	60.0	60.0
Annual Cost of Operation (millions)	\$255.9	\$270.0	\$248.7	\$242.9	\$245.4

Table 20-2 Estimated decommissioning costs for Zimmer Unit 1*

a.	Prompt dismantling and removal of nonradioactive structures	\$ 7.8
b.	Initial temporary entombment	\$11.0
c.	Total surveillance and maintenance for 104 years	\$ 9.9
d.	Dismantling and removal of remaining structure after 104 years of cooling	\$ 9.0
	Subtotal Decommissioning Cost	\$37.7
	Land Credit	(\$ 1.8)
	Net-Total Decommissioning Cost*	\$35.9

*All costs are stated in 1983 dollars.

reasonableness of the AIF decommissioning cost estimates, Battelle concluded that "[w]hile some of the individual cost calculations appear to be inconsistent and appear to omit certain cost items, the total costs do appear to be realistic."

In June 1980, the Pacific Northwest Laboratory operated by Battelle Memorial Institute published a detailed study entitled "Technology, Safety and Costs of Decommissioning a Referenced Boiling Water Reactor Power Station (NUREG/CR-0672). Using a large 1155-megawatt electric boiling water reactor power station as the basis for the study, the report concluded that decommissioning costs are estimated to be approximately $64.5 million in 1978 dollars for an initial entombment and deferred dismantlement decommissioning method. This cost estimate assumes a 100-year period between initial entombment and the starting of dismantlement. Although the applicants may realize some reduction in costs due to the smaller size of the Zimmer facility (792-megawatts electric versus the 1155-megawatt electric unit considered in the Battelle report), these cost savings are difficult to quantify and should not be significant. Nonetheless, some offset will occur to any potential savings when the above amount is restated in 1983 dollars. Since the above amount appears more conservative than the applicants' data as shown in Table 20-2, it has been adopted for use herein.

20.5 Financial Analysis - Sources of Funds

As indicated earlier, the Commission has interpreted the "reasonable assurance" requirement of financial qualification to be a "reasonable financing plan in light of relevant circumstances." Seabrook, 7 NRC 18 (1978). In consideration of the foregoing cost estimates, the following analysis will evaluate the reasonableness of the applicants' financial plans in covering the various amounts that will result from the operation of the facility.

In general, an evaluation of the financial plans of the applicants to meet operational expenses, TMI facility modification and operating costs, and decommissioning costs cannot be viewed in a vacuum but can only reasonably be considered in relation to other costs. Thus, the amounts to be financed must be considered in light of the operational characteristics and financial capabilities of the applicants. This financial perspective includes the applicants' nature of business, their size in revenue, assets, net income, and financial strength. Because the applicants are ongoing entities, such an evaluation requires a review of the financial results of their operation over a sustained period of time. As is the case with most financial reviews, emphasis is placed upon recent performance. The near-term financial outlook of the applicants is also given consideration. However, the near-term planning horizon is limited to the issue of how the projected costs of operation of the facility will fit into the general scheme of their operations.

Long-term financial considerations are also important in the financial review since some costs will occur over an extremely long time. However, the number of variables such as interest rates, the state of the stock and bond markets, inflation, and the cost of fuel and labor, among many others, makes long-term financial forecasting inherently uncertain. The operational characteristics of the applicants' financial condition gives a good indicator of their capabilities and therefore is important for long-term forecasts. In consideration of those relevant circumstances, the following evaluates the reasonableness of the applicants' financial plan.

20.6 Financing Plan - Zimmer Costs of Operation

The applicants plan to recover all costs of operation through revenues derived from their customers in their system-wide sales of electricity. Under the applicants' Joint Operation Agreement, the total costs of the facility's operation will be recovered in proportion to each applicant's ownership interests as follows: Cincinnati Gas and Electric Company - 40 percent, Columbus and Southern Ohio Electric Company - 28.5 percent, and Dayton Power and Light Company - 31.5 percent. By reason of rate regulation, their rates may only be increased upon approval by the Ohio Public Utility Commission.

Inherent in the operation of the Zimmer facility will be the production of electricity for the service of the applicants' customers. Because such capability will qualify the facility as a productive asset, from an accounting viewpoint such property will reasonably be expected to qualify as "property used and useful in public utility service."

As a consequence of this, the facility's cost of construction, including amounts allowed for funds used during construction, will be included in the rate base of each of the applicants for regulatory ratemaking purposes in the amount of their respective investments in it.* Under Ohio rate regulation, rate base inclusion of the facility will allow the applicants' recovery of the capital costs associated with its construction such as interest on debt, and dividends on preferred and common stock associated with and in addition to investment amounts formerly applied towards Zimmer's Construction. The same regulatory treatment also allows recovery of amounts associated with operation and maintenance expenses necessary for the production of power that will be used by the applicants' customers. These amounts normally include all reasonable fixed and variable costs including the return of the original investment in the form of depreciation.

Since the applicants have demonstrated an historically consistent recovery of such amounts in capital and operating costs for all other significant facilities they have formerly constructed and have both formerly and presently operated, it is reasonable to conclude that the applicants' plan to finance the facility's operation through internally generated amounts in the form of revenues derived from rates charged to customers for utility service, both produced overall and especially that generated by the Zimmer facility, represents a reasonable financing plan in light of relevant circumstances.

20.7 Financing Plan - Decommissioning Costs

The applicants presently plan to obtain the funds required for both intermediate and ultimate decommissioning of the plant through revenues derived from their customers to reflect annual depreciation charges during the service life of the facility to produce amounts which will be deposited with a trustee. The sum of the amounts so deposited plus earnings accruing thereon during the operating life of the facility would be the source of funds to meet decommissioning

*Each of the applicants is presently allowed approximately 50 percent of the Zimmer facility's construction costs as a part of their present rate bases in their last ratemaking proceedings before the Public Utilities Commission of Ohio.

costs. Based on a 6.5 percent annual inflation rate from 1983 through the final dismantlment/removal of the remaining structures in the year 2120 and a 6 percent tax-free interest rate on funds deposited with the decommissioning trustee, the annual payments to the decommissioning trustee over the 33-year plant life required to provide the necessary funds for each of the four components of decommissioning along with the land credit as well as the total annual payment are shown in Table 20-3.

Table 20-3 Annual amounts to be deposited with decommissioning trustee over the 33-year operating life required to provide estimated necessary funds for decommissioning*

a. Prompt dismantling and removal of nonradioactive structures	$ 639,874
b. Initial temporary entombment	$ 903,373
c. 104 years of surveillance and maintenance	$1,041,489
d. Dismantling and removal of remaining structures after 104 years of cooling	$1,209,490
Subtotal annual decommissioning fund deposited over 33-year operating lifetime	$3,794,227
Land Credit	($ 242,617)
Net-Total annual decommissioning fund deposited over 33-year operating lifetime	$3,551,610

*The estimated annual payments shown above are based on the decommissioning cost estimates shown in Table 2, a 6.5 percent annual inflation rate after 1983, and a 6 percent tax-free interest rate on funds deposited with the decommissioning trustee.

By utilizing the assumptions made by the applicants, the staff has verified the computation of the sinking fund amounts stated by the applicants above in Table 20-3.

As stated earlier, each of the applicants intends to reflect its respective proportionate share of estimated decommissioning costs as a component of its annual depreciation expenses to be attributable to operating the Zimmer facility in operation. Using the ownership interest proportions of the applicants in the Zimmer facility, the total annual required payments for each of the applicants to meet their estimated $3.6 million annual payment to accrue $35.9 million of total decommissioning expenses in 1983 dollars is shown in Table 20-4 below.

Table 20-4 Annual payments required by the Zimmer facility co-owners to meet decommissioning expenses of $35.9 million through a 33-year annual sinking fund payment of $3.6 million

Facility Co-Owner	Annual Payment Amount (Millions)
Cincinnati Gas & Electric Company	$1.44
Columbus and Southern Ohio Electric Company	1.03
Dayton Power and Light Company	1.13
Total	$3.60

As stated earlier, however, this analysis bases total estimated decommissioning costs upon the Battelle Pacific Northwest Laboratories projection of $64.5 million. By utilization of a sinking fund method of financing under a 6 percent tax-free yield accruing on all amounts so deposited, the annual aggregate payment requirement increases to $4.53 million. The required payment amounts for each of the applicants to meet the above decommissioning expense estimate is shown in Table 20-5 below.

Table 20-5 Annual payments required by the Zimmer facility co-owners to meet decommissioning expenses of $64.5 million through a 33-year annual sinking fund payment of $4.5 million

Facility Co-Owner	Annual Payment Amount (Millions)
Cincinnati Gas & Electric Company	$1.8
Columbus and Southern Ohio Electric Company	1.3
Dayton Power and Light Company	$1.4
Total	$4.5

Under a sinking fund method of financing, the sum of the amounts annually deposited with tax-free interest earnings accruing over the operating life of the facility appreciate to meet the estimated future costs to decommission the Zimmer facility. Since these amounts are proposed to be derived through revenues as a recovery of costs for each of the applicants during the operating life of the facility, they must be approved by the Public Service Commission of Ohio. Approval for the inclusion of such amounts in revenues charged to customers for utility service requires that they be used for the production of the service. Since the NRC requires that any operating reactor be safely decommissioned when ret'red, it is reasonable to assume that those amounts are necessary and rea. onable expenses, especially when they are also incurred for the protection of the public health and safety. Accordingly, because decommissioning costs will be a necessary component in the providing of utility service from the Zimmer facility, it is reasonable to believe that the applicants' plan to finance these expenses from customer revenues through approval by the Public Utility Commission of Ohio constitutes a reasonable financing plan in light of relevant circumstances.

Since the co-owner applicants will be sharing all costs arising out of the operation of the Zimmer facility, the impact of any cost increases will be shared among them. The relative magnitude of any such cost increase impact may be seen from comparison of Table 20-4 and 20-5 above. Under a $30 million increase in decommissioning expenses, the co-owner having the largest interest in the Zimmer facility - Cincinnati Gas and Electric at 40 percent - would need only an additional $400,000 annual payment requirement to its share of the tax-free sinking fund. From a relative viewpoint, this would cause less than one-tenth of one percent dollar impact to its 1979 gross electric operating revenues of $518.9 million and less than one half of one-tenth of one percent to its total operating revenues for 1979 of $825.8 million. The impact of any such increase in decommissioning costs would be similar for both Dayton Power and Light Company which realized total annual operating revenues of $535.8 million in 1979 and Columbus and Southern Ohio Electric Company which realized $416.8 million.

Moreover, although the NRC requires no specific plan to fund decommissioning expenses, the staff believes that the applicants' plan to fund such amounts in an independent tax-free investment vehicle with a trustee provides an additional element of assurance in that it constitutes an especially liquid method for obtaining the necessary amounts of proceeds to meet decommissioning costs. Because there is always a ready market for tax-free securities, there will be little difficultly in liquidating the investment when the need arises. Furthermore, should additional amounts be needed over and above those invested accrued in the tax-free trustee account, the applicants have two other traditional sources of funds available to meet any such expenses. The first source is the applicants' internal cash generation attributable to: (1) depreciation expenses for all utility plant; (2) retained earnings; and (3) normalized tax depreciation and levelized investment tax credits. These are noncash expenses which utilities normally recover through revenues to meet their capital requirements on an internal basis. The second source of funds is the external capital market. As public utilities constitute the most capital-intensive industry in the United States, they have long had access to funds in the public securities market. To access such additional external funds, the applicants would issue debt in the form of bonds or issue additional preferred or common stock, or a combination of each. A three-year summary of each of the applicants' recent internal and external financings is shown in Table 20-6 below.

Table 20-6 Summary of Zimmer facility applicants' internal and external financings for the period 1977 to 1979 (dollars in millions)

Facility Co-Owners	Year 1977	1978	1979
Cincinnati Gas & Electric Company			
Internal Financings	$ 58.6	$ 56.6	$ 54.6
External Financings	$ 33.7	$147.3	$205.2
Columbus and Southern Ohio Electric Company			
Internal Financings	$ 31.3	$ 6.4	$ 44.3
External Financings	$114.1	$106.8	$ 59.0
Dayton Power and Light Company			
Internal Financings	$ 32.3	$ 27.9	$ 23.8
External Financings	$ 81.5	$113.8	$176.1

The historic ability of the applicants' successful access to these markets provides an even further degree of assurance that the necessary funds will be available to meet decommissioning costs when necessary.

20.8 Conclusion

In accordance with the regulations cited herein, an applicant must demonstrate that it has reasonable assurance of obtaining the necessary funds to cover the estimated costs of the activities contemplated under the license. As stated earlier, the Commission has determined in Seabrook that the reasonable assurance requirement for financial qualifications is a reasonable financing plan in light of relevant circumstances. Based upon the preceding analyses of their proposed financing plans, the staff concludes that Cincinnati Gas and Electric Company Columbus and Southern Ohio Edison Company, and Dayton Power and Light Company have reasonable financing plans in light of relevant circumstances to operate, shutdown, if necessary, and maintain the Zimmer facility in a safe condition.

Accordingly, the staff has determined that the applicants have a reasonable assurance to obtain the estimated funds necessary to perform the activities contemplated by the applicants under the proposed operating license to the extent of their ownership interest in the facility. As a consequence of this, the staff finds that the applicants are financially qualified to operate and safely decommission the Zimmer Nuclear Power Station Unit No. 1. In summary, our conclusion is based upon the applicants' status as regulated public utilities, the size of their operations, their demonstrated ability to achieve revenues sufficient to cover each of their operating and capital costs, and their successful history of obtaining capital in amounts both internally generated and in the external markets.

21 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

21.1 General

Pursuant to the financial protection and indemnification provisions of the Atomic Energy Act of 1954, as amended (Section 170 and related sections), the Commission has issued regulations in 10 CFR Part 140. These regulations set forth the Commission's requirements with regard to proof of financial protection by, and indemnification of, licenses for facilities such as power reactors under 10 CFR Part 50.

21.2 Preoperational Storage of Nuclear Fuel

The Commission's regulations in 10 CFR Part 140 require that each holder of a construction permit under 10 CFR Part 50, who is also the holder of a license under 10 CFR, Part 70 authorizing the ownership and possession for storage only of special nuclear material at the reactor construction site for future use as fuel in the reactor (after issuance of an operating license under 10 CFR Part 50), shall, during the interim storage period prior to licensed operation, have and maintain financial protection in the amount of $1,000,000 and execute an indemnity agreement, with the Commission. Proof of financial protection is to be furnished prior to, and the indemnity agreement executed as of, the effective date of the 10 CFR Part 70 license. Payment of an annual indemnity fee is required.

The applicant has furnished the Commission proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association Policy (Nuclear Energy Liability Policy, facility form No. NF-210). Further, the applicant has executed an indemnity agreement with the Commission effective as of the date of its preoperational fuel storage license. The applicant has paid the annual indemnity fee applicable to preoperational fuel storage.

21.3 Operating Licenses

Under the Commission's regulations, 10 CFR Part 140, a license authorizing the operation of a reactor may not be issued until proof of financial protection in the amount required for such operation has been furnished, and an indemnity agreement covering such operation (as distinguished from preoperational fuel storage only) has been executed. The amount of financial protection which must be maintained for William H. Zimmer, Unit 1 (which has a rated capacity in excess of 100,000 electric kilowatts), is the maximum amount available from private sources, which is currently $520 million.

Accordingly, licenses authorizing operation of William H. Zimmer, Unit 1, will not be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

We expect that, in accordance with the usual procedure, the nuclear liability insurance pools will provide, several days in advance of anticipated issuance of the operating license document, evidence in writing on behalf of the applicant, that the present coverage has been appropriately amended so that the policy limits have been increased, to meet the requirements of the Commission's regulations for reactor operation. Similarly, operating licenses will not be issued until an appropriate amendment to the present indemnity agreement has been executed. The applicant will be required to pay an annual fee for operating license indemnity as provided in the Nuclear Regulatory Commission's regulations, at the rate of $6 per thousand kilowatts of thermal capacity authorized in his operating license. On the basis of the above considerations, we conclude that the presently applicable requirements of 10 CFR Part 140 have been satisfied and that, prior to issuance of the operating licenses, the applicant will be required to comply with the provisions of 10 CFR Part 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and as to the execution of an appropriate indemnity agreement with the Commission.

22 TMI-2 REQUIREMENTS

22.1 Introduction

The accident at Three Mile Island (TMI) Unit 2 resulted in requirements which were developed from the recommendations of several groups established to investigate the accident. These groups include the Congress, the General Accounting Office, the President's Commission on the accident at Three Mile Island, the Nuclear Regulatory Commission (NRC) Special Inquiry Group, the NRC Advisory Committee on Reactor Safeguards (ACRS), the Lessons-Learned Task Force and the Bulletins and Orders Task Force of the NRC Office of Nuclear Reactor Regulation, the Special Review Group of the NRC Office of Inspection and Enforcement, the NRC Staff Siting Task Force and Emergency Preparedness Task Force, and the NRC Offices of Standards Development and Nuclear Regulatory Research. The report NUREG-0660 entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident" (Action Plan) was developed to provide a comprehensive and integrated plan for the actions now judged necessary by the NRC to correct or improve the regulation and operation of nuclear facilities. The Action Plan was based on the experience from the TMI-2 accident and the recommendations of the investigating groups.

In the development of the Action Plan (NUREG-0660), the NRC has transformed the recommendations of the investigating groups into discrete scheduled tasks that specify changes in its regulatory requirements, organization, or procedures. Some actions to improve the safety of operating plants were judged to be necessary before an action plan could be developed, although they were subsequently included in the Action Plan. Such actions came from the Bulletins and Orders issued by the Commission immediately after the accident, the first report of the Lessons-Learned Task Force issued in July 1979, and the recommendations of the Emergency Preparedness Task Force. Before these, immediate actions were applied to operating plants they were approved by the Commission.

Our review of TMI-2 requirements is based on the Commission's guidance issued on June 16, 1980, regarding the requirements to be met for current operating license applications. The requirements are derived from NRC's Action Plan (NUREG-0660) and are found in NUREG-0694, "TMI-Related Requirements for New Operating License," and NUREG-0737, "Clarification of TMI Action Plan Requirements." Zimmer was measured against the NRC regulations as augmented by these requirements.

Section 22 of this report addresses the applicant's implementation of the TMI-related requirements in Zimmer. In the Final Safety Analysis Report, the applicant has provided its response to our requirements. The items in Section 22.2 correspond to the items designated in NUREG-0737. During our review, we met with the applicant in Bethesda and at the Zimmer site. The applicant has amended its initial response as a result of our review. Meeting results and applicant's letter relevant to our review are discussed in applicable sections of this supplement.

22.2 TMI Action Plan Requirements for Applicants for Operating Licenses

I. Operational Safety

I.A.1.1 Shift Technical Advisor

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multi-unit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

There are no changes to the previous requirements resulting from NUREG-0660 and the October 30, 1979 letter from H. R. Denton to all operating nuclear power plants.

Clarification

The letter of October 30, 1979 clarified the short-term STA requirements. That letter indicated that the STAs must have completed all training by January 1, 1981. The following clarification was provided in NUREG-0737, dated October 1980.

The need for the STA position may be eliminated when the qualifications of the shift supervisors and senior operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until these long-term improvements are attained, the need for an STA program will continue.

The staff has not yet established the detailed elements of the academic and training requirements of the STA beyond the guidance given in its October 30, 1979 letter. Nor has the staff made a decision on the level of upgrading required for licensed operating personnel and the man-machine interface in the control room that would be acceptable for eliminating the need of an STA. Until these requirements for eliminating the STA position have been established, the staff continues to require that, in addition to the staffing requirements specified in its July 31, 1980 letter (as revised by Item I.A.1.3 of this enclosure), an STA be available for duty on each operating shift when a plant is being operated in Modes 1-4 for a PWR and Modes 1-3 for a BWR. At other times, an STA is not required to be on duty.

Since the October 30, 1979 letter was issued, several efforts have been made to establish, for the longer term, the minimum level of experience, education, and training for STAs. These efforts include work on the revision to ANS-3.1., work by the institute of Nuclear Power Operations (INPO), and internal staff efforts.

INPO has made available a document dated April 30, 1980, entitled "Nuclear Power Plant Shift Technical Advisor--Recommendations for Position Description, Qualifications, Education and Training." Sections 5 and 6 of the INPO document describe the education, training, and experience requirements for STAs. The NRC staff finds that the descriptions as set forth in Sections 5 and 6 of Revision 0 to the INPO document are an acceptable approach for the selection and training of personnel to staff the STA positions. (Note: This should not be interpreted to mean that this is an NRC requirement at this time. The intent is to refer to the INPO document as acceptable for interim guidance for a utility in planning its STA program over the long term (i.e., beyond the January 1, 1981 requirement to have STAs in place in accordance with the qualification requirements specified in the staff's October 30, 1979 letter).)

No later than January 1, 1981, all licensees of operating reactors shall provide this office with a description of their STA training program and their plans for requalification training. This description shall indicate the level of training attained by STAs by January 1, 1981 and demonstrate conformance with the qualification and training requirements in the October 30, 1979 letter. Applicants for operating licenses shall provide the same information in their application, or amendments thereto, on a schedule consistent with the NRC licensing review schedule.

Discussion and Conclusions

The applicant is training approximately 20 staff engineers in order to qualify them to serve as shift technical advisors (STAs). These engineers, when qualified, will serve as on-duty STAs for 24 hours at a time and on a rotating basis such that each individual will serve a 24 hour tour of duty as STA approximately once every 2 weeks.

An STA will be on duty at all times when the unit is in power operation, startup, or hot shutdown (operating modes 1, 2 or 3) and will be available to the on-duty shift supervisor and to the control room within 10 minutes whenever summoned. The applicant has stated in its April 22, 1981 submittal that:

> "During a typical 24-hour tour, the duty STA, after appropriate relief, will spend the first portion of his/her day in the control room to assure that he/she has a firm grasp, understanding and awareness of plant status, conditions and activities. The STA will interface with the shift supervisor, licensed and non-licensed plant operators; maintain an awareness of and participate in surveillance tests and other activities; be present and observant during all shift reliefs during his/her duty assignment; and have no concurrent or additionally assigned operating or maintenance duties for the 24-hour period. Since the onsite STA will be in a duty status for longer than one shift, he/she will be asleep at times. The onsite STA sleeping facilities are within 2 minutes comfortable walking distance from the main control room, assuring the 10-minute availability requirement. Periodic plant tours, coupled with periodic control room checks, will serve to keep the STA cognizant of plant status. During the back shifts, the STA may be assigned responsibilities such as review of operating experience

reports, investigating events or review of plant procedures, systems, design, etc., that could have a potential for improving plant safety. The STAs will interface with members of the ISEG to enhance that safety group's contact with and knowledge of day-to-day plant operation. Additionally, during third shift operation, the STA will return to the control room at least every 2 hours to assure he/she is aware of plant status and any evolutions being carried out or scheduled to be carried out.

When not actually serving a 24-hour tour of duty as STA, these STA qualified engineers work in their various support staff jobs at the plant. The initial group of engineers presently receiving STA training includes the following personnel:

station chemist
health physics engineer
instrument and controls supervisor
maintenance supervisor
maintenance staff engineer
operations staff engineer
technical engineer
reactor engineer
station staff engineers
station quality engineer
assistant quality engineer
nuclear engineers

The applicant has stated that all of its present group of engineers in training for STA service will meet the staff's requirements for STA education and training qualifications as delineated in the NRC's letter to licensees dated October 30, 1979, prior to fuel load. We have reviewed the training program proposed by the applicant for providing additional college level education and for providing training in Zimmer station systems and operations and accident analysis. This program is being carried out through the University of Cincinnati. Based on this review, we conclude that the STA candidates that receive this training will meet the current minimum NRC requirements for STA education and training qualifications as described in the October 30, 1979 letter to licensees. The applicant has also informed us that it has implemented a longer range training program for its STAs that is based on the guidance furnished by the Institute of Nuclear Power Operations document "Nuclear Power Plant Shift Technical Advisor--Recommendations for Position Description, Qualifications, Education and Training" which was endorsed as interim guidance in NUREG-0737, "Clarification of TMI Action Plan Requirements."

The NRC requirements published to date do not preclude the 24-hour tour of duty as a way to provide STA coverage. In fact, the clarification of the staff position on STAs as provided in the NRC's October 30, 1979 letter to licensees stated "The onsite STA may be in a duty status for periods of time longer than one shift and, therefore, asleep at some times, if 10-minute availability is assured."

The use of STAs on a 24-hour rotating basis provides several possible advantages over the more "conventional" assignment of personnel to specific rotating shifts of 8-hour duration:

1. Highly motivated, trained and experienced technical personnel who are unwilling or unavailable to work on a conventional rotating shift assignment are willing and available to work the infrequent 24-hour tour of STA duty. It has been the applicant's experience thus far that quality engineering personnel do not choose or desire permanent shift assignments. It appears that it will be even more difficult to attract and maintain this type of person in the future unless concurrent tasks, responsibilities and challenges are factored into the work assignments.

2. More technical support staff personnel are exposed to operational problems and situations with greater cross fertilization of support staff and operations staff experiences due to the interaction between operating personnel and the many different STAs. An understanding of needs and problems from several perspectives enhances safety as well as cooperation in emergency situations.

3. The use of the large pool of STA qualified personnel allows new professional employees, after appropriate training and education, to integrate into the STA duty rotation. At the same time, the program provides highly trained personnel to accumulate valuable operational experience that benefits the entire nuclear operation at all levels.

4. The use of the large pool of STA qualified personnel eases the problems created due to unanticipated personnel losses, sickness, or vacation. It will also allow an STA who performs unsatisfactorily to be removed from the STA pool with little impact on the STA program.

There are also possible disadvantages of the 24-hour tours and the large cadre of STAs as compared to the system whereby STAs serve with specific rotating shift crews:

1. An attitude could develop due to the infrequent duty wherein an individual could become complacent because an emergency would not be as likely to occur when duty assignments rotate on an approximate 2-week schedule with the result that the STAs might not pay adequate attention to their training lessons and assignments or other information disseminated to them through the feedback system.

2. The STAs under the proposed system are dispersed throughout the plant support staff and report to many different supervisors in their normal job junction. This may result in administrative problems such as a lack of uniform direction being given to STAs concerning their STA activities. Information that should be given to STAs as part of operating experience feedback may not be adequately disseminated to them. Retraining and new training for STAs may become a scheduling problem unless these individuals serving as STAs have specific training weeks assigned to them as does the operating shift crew.

3. Due to the infrequency with which the STA will work with any particular shift crew, it is less likely that a team working relationship will develop between the STA and the shift crew. A team relationship would tend to enhance the coordination of the accident response, while lack of such working relationship could tend to diminish such coordination. (Note: An argument in favor of not having a close working relationship between

the STA and the shift crew is that such close relationships may reduce the independence of the STA's recommendations and hence reduce their value.)

4. Due to the infrequency with which each individual is assigned to STA duties, they are less likely to develop the operating and system knowledge and expertise that they would develop if they worked every day as an STA. They are also less likely to be as aware of the details of current operational problems or activities.

It is our opinion that the circadian rhythm upsets caused by rotating shift work would produce greater fatigue during a single 8-hour shift (evenings or midnights) than would be caused during a single 24-hour tour of duty as described above. However, because we are unable to scientifically prove this to be so, we did not assign a fatigue factor advantage to the 24-hour tour of duty.

We believe that the use of STAs as proposed by the applicant could result in more highly experienced, trained and motivated technical personnel being assigned as STAs at the Zimmer Station than would otherwise be the case. We also believe that this and the other possible advantages of the proposed STA program as discussed above are substantial and outweigh all of the disadvantages that we have identified. On this basis, we conclude that the applicant's proposal to use a large number of fully trained STAs such that they are assigned to 24-hour tours of duty as STA approximately once every 2 weeks or less is acceptable.

I.A.1.2 Shift Supervisor Administrative Duties

Position

Review the administrative duties of the shift supervisor and delegate functions that detract from or are subordinate to the management responsibility for assuring safe operation of the plant to other personnel not on duty in the control room.

Clarification

1. The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the shift supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

2. Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the shift supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

   a. The responsibility and authority of the shift supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The principle shall be reinforced that the shift supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

b. The shift supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of control room operators. Persons authorized to relieve the shift supervisor shall be specified.

c. If the shift supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authority shall be clearly specified.

3. Training programs for shift supervisors shall emphasize and reinforce the responsibility for safe operation and the management function that the shift supervisor is to provide for assuring safety.

4. The administrative duties of the shift supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

Discussion and Conclusions

The applicant committed in its April 22, 1981 submittal to issue a directive, over the signature of the Senior Vice President, to all Zimmer Station personnel that clearly establishes the shift supervisor's primary management responsibility for safe operation of the plant and that clearly establishes the shift supervisor's command duties. This directive will be issued prior to fuel load and will be reissued annually.

The applicant also committed that the Senior Vice President will review the administrative duties of the shift supervisor and assure that any that detract from or are subordinate to the management responsibility for safe operation of the plant are delegated to other personnel not on duty in the control room. This will be done prior to fuel load.

The shift supervisor's duties are delineated in Zimmer Station Administrative Directive OS-SAD-01. Revision 7 of this directive describes the duties, responsibilities and authority of the shift supervisor as required by Items 2a, and 2b of the above NRC position. It does not, however, clearly designate, as required by Item 2c above, that a licensed senior reactor operator (a senior nuclear control operator in the applicant's terminology) will be designated to assume the control room function if the shift supervisor is absent from the control room when the plant is in operating modes 1, 2, or 3. The NRC requires that a licensed senior reactor operator be present in the control room at all times whenever the unit is in operating modes 1, 2, or 3. The applicant has agreed to modify the directive to reflect this requirement prior to fuel load.

The applicant has committed to augment its training program to provide appropriate training for the shift supervisors that emphasizes and reinforces the shift supervisor's command duties and management responsibility for safe operation of the plant. We conclude that this conforms with Item 3 of the above position and is acceptable.

The NRC Office of Inspection and Enforcement will review the management directive and the administrative procedures and records related to review of the shift supervisor's administrative duties to assure that the above commitments have been fulfilled to conform with the above NRC position Items 1, 2C and 4 prior to fuel load. Subject to this confirmation, we conclude that the administrative procedures meet Action Plan Item I.A.1.2 requirements and are acceptable.

I.A.1.3 Shift Manning

Position

Assure that the necessary number and availability of personnel to man the operations shifts have been designated by the licensee. Administrative procedures should be written to govern the movement of key individuals about the plant to assure that qualified individuals are readily available in the event of an abnormal or emergency situation. This should consider the recommendations on overtime in NUREG-0578. Provisions should be made for an aide to the shift supervisor to assure that, over the long term, the shift supervisor is free of routine administrative duties.

Clarification

At any time a licensed nuclear unit is being operated in Modes 1-4 for a pressurized water reactor (power operation, startup, hot standby or hot shutdown, respectively) or in Modes 1-3 for a boiling water reactor (power operation, startup, or hot shutdown, respectively), the minimum shift crew shall include two licensed senior reactor operators, one of whom shall be designated as the shift supervisor, two licensed reactor operators, and two unlicensed auxiliary operators. For a multi-unit station, depending upon the station configuration, shift staffing may be adjusted to allow credit for licensed senior reactor operators and licensed reactor operators to serve as relief operators on more than one unit; however, these individuals must be properly licensed on each such unit. At all other times, for a unit loaded with fuel, the minimum shift crew shall include one shift supervisor who shall be a licensed senior reactor operator, one licensed reactor operator, and one unlicensed auxiliary operator.

Adjunct requirements to the shift staffing criteria stated above are as follows:

1. A shift supervisor with a senior reactor operator's license, who is also a member of the station supervisory staff, shall be onsite at all times when at least one unit is loaded with fuel.

2. A licensed senior reactor operator shall, at all times, be in the control room from which a reactor is being operated. The shift supervisor may, from time to time, act as relief operator for the licensed senior reactor operator assigned to the control room.

3. For any station with more than one reactor containing fuel, the number of licensed senior reactor operators onsite shall, at all times, be at least one more than the number of control rooms from which the reactors are being operated.

4. In addition to the licensed senior reactor operators specified in (1), (2), and (3) above, for each reactor containing fuel, a licensed reactor operator shall be in the control room at all times.

5. In addition to the operators specified in (1), (2), (3), and (4) above, for each control room from which a reactor is being operated, an additional licensed reactor operator shall be onsite at all times and available to serve as relief operator for that control room. As noted above, this individual may serve as relief operator for each unit being operated from that control room, provided he holds a current license for each unit.

6. Auxiliary (non-licensed) operators shall be properly qualified to support the unit to which assigned.

7. In addition to the staffing requirements stated above, shift crew assignments during periods of core alterations shall include a licensed senior reactor operator to directly supervise the core alterations. This licensed senior reactor operator may have fuel handling duties but shall not have other concurrent operational duties.

Licensees of operating plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the plant. These provisions are required to assure that qualified plant personnel to man the operational shifts are readily available in the event of an abnormal or emergency situation.

These administrative procedures shall also set forth a policy, the objective of which is to operate the plant with the required staff and develop working schedules such that use of overtime is avoided, to the extent practicable, for the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, instrumentation and control technicians, and key maintenance personnel).

IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," dated February 1, 1980, discusses the concern of overtime work for members of the plant staff who perform safety-related functions.

We recognize that there are diverse opinions on the amount of overtime that would be considered permissible and that there is a lack of hard data on the effects of overtime beyond the generally recognized normal 8-hour working day, the effects of shift rotation, and other factors. We have initiated studies in this area. Until a firmer basis is developed on working hours, the administrative procedures shall include as an interim measure the following guidance, which generally follows that of IE Circular No. 80-02.

In the event that overtime must be used (excluding extended periods of shutdown for refueling, major maintenance, or major plant modifications), the following overtime restrictions should be followed.

1. An individual should not be permitted to work more than 12 hours straight (not including shift turnover time).

2. There should be a break of at least 12 hours (which can include shift turnover time) between all work periods.

3. An individual should not work more than 72 hours in any 7-day period.

4. An individual should not be required to work more than 14 consecutive days without having 2 consecutive days off.

However, recognizing that circumstances may arise requiring deviation from the above restrictions, such deviation shall be authorized by the plant manager or his deputy or higher levels of management in accordance with published procedures and with appropriate documentation of the cause. If a reactor operator or senior reactor operator has been working more than 12 hours during periods of extended shutdown (e.g., at duties away from the control board), such individuals shall not be assigned shift duty in the control room without at least a 12-hour break preceding such an assignment. We encourage the development of a staffing policy that would permit the licensed reactor operators and senior reactor operators to be periodically relieved of primary duties at the control board, such that periods of duty at the board do not exceed about 4 hours at a time. If a reactor operator is required to work in excess of 8 continuous hours, he shall be periodically relieved of primary duties at the control board, such that periods of duty at the board do not exceed about 4 hours at a time.

The guidelines on overtime do not apply to the shift technical advisor provided he or she is provided sleeping accommodations and a 10-minute availability is assured.

Operating license applicants shall complete these administrative procedures before fuel loading. Development and implementation of the administrative procedures at operating plants will be reviewed by the Office of Inspection and Enforcement beginning 90 days after July 31, 1980.

Discussion and Conclusion

The applicant has informed us of its shift manning and staffing plans in its submittal dated April 22, 1981. The applicant plans to provide six qualified operating shift crews beginning with the initial fuel loading and continuing thereafter. This will allow each of the shift crews to be assigned to requalification training every sixth shift rotation.

The applicant stated that it has currently prepared 18 shift personnel for senior reactor operator license examinations and eight shift personnel for reactor operator license examinations. It has also currently prepared eight additional management personnel for senior reactor license examination. In addition, it expects to have two more shift personnel prepared for license examination by fuel load to, as a minimum, supplement the reactor operator group.

Further, the applicant has indicated that it has seven additional reactor operator candidates that will have completed preparation for and taken the hot license examination within 3 months following fuel load. It has also indicated that it will have 21 non-licensed plant operators available for shift duty at fuel load.

Assuming that approximately 20 percent of the candidates for licenses do not pass the examinations, we estimate that the applicant will have approximately 16 licensed senior reactor operators and eight licensed reactor operators available to man the shift crews starting at fuel load and approximately five more licensed reactor operators three months later. Thus, we estimate that there will be a sufficient number of licensed and non-licensed shift personnel to man six shift crews at fuel load unless some of them resign between now and fuel load. If there are resignations of licensed personnel, the number of shift crews available would be reduced until the additional operator candidates receive licenses approximately three months after fuel load.

The training and staffing plan submitted by the applicant covers the period from the present time until 2 years following fuel load. The submitted plan is based on an assumed attrition rate due to examination failures, resignation, promotion, and all other causes of 14 percent per year. However, the applicant has stated that it will adjust the hiring and training to maintain the six shift crews fully staffed with whatever attrition rate actually occurs.

On the basis of our review as discussed above, we find that the applicant's staffing plans can be expected to provide sufficient licensed and non-licensed operators to staff the plant without a need for routine overtime. We conclude that the shift manning for the plant, subject to the acceptable completion of the NRC operator licensing examinations, as discussed above, meets the staffing requirements of Action Plan Item I.A.1.3 as stated above and is acceptable.

Zimmer Station Administrative Directive OS.SAD.01 addresses the overtime work limitations for plant operations. Revision 7 of this directive addresses the overtime requirements as presented in the NRC's July 31, 1980 letter to licensees but does not address the revisions to these overtime requirements as subsequently promulgated by NUREG-0737. Following discussion with us on this subject, the applicant has indicated in its April 22, 1981 submittal that it is now addressing these NUREG-0737 revisions to the overtime requirements in its station administrative directives. Based on our review of the April 22, 1981 submittal, we find that the modifications that the applicant has committed to make are in conformance with the NUREG-0737 overtime requirements except in the area of authorization (inferred to mean prior authorization) by the plant manager for overtime work deviating from the restriction limits. The applicant stated that overtime work deviations would be reviewed and documented following rather than prior to performance of the work.

We have informed the applicant that we will require that deviations for the overtime limits by personnel performing safety related functions as distinguished below be reviewed and authorized prior to performing the deviating work.

In the case of deviations from the overtime limits by on-duty operating shift crew members or the on-duty health physics technician, all deviations must be reviewed prior to performing the work and authorized by the plant manager or his designee (someone normally designated to act for the plant manager such as the assisstant superintendent or the on-duty emergency director).

In the case of deviations from the overtime limits by other personnel performing safety related functions, (e.g., maintenance personnel), all preplanned or scheduled and posted work that deviates from the limits must be reviewed prior to performing the work by the group supervisor (e.g., maintenance engineer) or the plant manager or his designee. However, for this latter group of "other personnel" performing safety related functions and in case of unforeseen shift-to-shift contingencies and emergencies, deviations to the overtime limits may be reviewed and authorized on shift by the group foreman prior to performance of the work with subsequent review by the appropriate group supervisor or the plant manager.

The applicant has agreed to modify the station administrative directives to incorporate these requirements.

Subject to the incorporation of the overtime limit requirements as discussed above, we conclude that the applicant's administrative directives concerning overtime limits are in accordance with the guidance of Action Plan Item I.A.1.3 and are acceptable.

The NRC Office of Inspection and Enforcement will review the station directives to assure that they are modified as stated above and implemented prior to fuel load.

I.A.2.1 Immediate Upgrading of Operator and Senior Operator Training and Qualification

Position

Applicants for Senior Reactor Operator (SRO) license shall have 4 years of responsible power plant experience, of which at least 2 years shall be nuclear power plant experience (including 6 months at specific plant) and no more than 2 years shall be academic or related technical training. After fuel loading applicants shall have 1 year of experience as a licensed operator or equivalent.

Certifications that operator license applicants have learned to operate the controls shall be signed by the highest level of corporate management for plant operation.

Applicants must revise training programs to include training in heat transfer, fluid flow, thermodynamics, and plant transients.

Clarification

Applicants for SRO either come through the operations chain (C operator to B operator to A operator, etc.) or are degree-holding staff engineers who obtain licenses for backup purposes.

In the past, many individuals who came through the operator ranks were administered SRO examinations without first being an operator. This was clearly a poor practice and the letter of March 28, 1980 requires reactor operator experience for SRO applicants.

However, NRC does not wish to discourage staff engineers from becoming licensed SROs. This effort is encouraged because it forces engineers to broaden their knowledge about the plant and its operation.

In addition, in order to attract degree-holding engineers to consider the shift supervisor's job as part of their career development, NRC should provide an alternate path to holding an operator's license for 1 year.

The track followed by a high-school graduate (a nondegreed individual) to become an SRO would be 4 years as a control room operator, at least one of which would be as a licensed operator, and participation in an SRO training program that includes 3 months on shift as an extra person.

The track followed by a degree-holding engineer would be, at a minimum, 2 years of responsible nuclear power plant experience as a staff engineer, participation in an SRO training program equivalent to a cold applicant training program, and 3 months on shift as an extra person in training for an SRO position.

Holding these positions assures that individuals who will direct the license activities of licensed operators have had the necessary combination of education, training, and actual operating experience prior to assuming a supervisory role at that facility.

The staff realizes that the necessary knowledge and experience can be gained in a variety of ways. Consequently, credit for equivalent experience should be given to applicants for SRO licenses.

Applicants for SRO licenses at a facility may obtain their 1-year operating experience in a licensed capacity (operator or senior operator) at another nuclear power plant. In addition, actual operating experience in a position that is equivalent to a licensed operator or senior operator at military propulsion reactors will be acceptable on a one-for-one basis. Individual applicants must document this experience in their individual applications in sufficient detail so that the staff can make a finding regarding equivalency.

Applicants for SRO licenses who possess a degree in engineering or applicable sciences are deemed to meet the above requirements, provided they meet the requirements set forth in Sections A.1.a and A.2 in enclosure 1 in the letter from H. R. Denton to all power reactor applicants and licensees, dated March 28, 1980, and have participated in a training program equivalent to that of a cold senior operator applicant.

NRC has not imposed the 1-year experience requirement on cold applicants for SRO licenses. Cold applicants are to work on a facility not yet in operation; their training programs are designed to supply the equivalent of the experience not available to them.

Discussion and Conclusions

The above requirements have been implemented by the applicant at Zimmer Nuclear Power Station effective with all submittals for operator and senior operator licenses. Individual applicants shall be reviewed to assure that they continue to meet the above requirements. If it is necessary to deviate from the prerequisite levels of experience for SRO's, the duration shall be identified and justified on individual license applications by the applicant.

Certifications that operator license applicants have learned to operate the controls will be signed by the highest level of corporate management for plant operation.

The Cincinati Gas and Electric Company has submitted a revised training program that includes training in areas required by the action plan for I.A.2.1. The training programs in heat transfer, fluid flow and thermodynamics have been developed and are presently taught by the applicant.

We conclude that the Cincinati Gas and Electric Company has satisfied the requirements of this task of the action plan.

I.A.2.3 Administration of Training Programs for Licensed Operators

Position

Pending accreditation of training institutions, training instructors who teach systems, integrated response, transient and simulator courses shall successfully complete a Senior Reactor Operator (SRO) examination prior to fuel loading and instructors shall attend appropriate retraining programs that address, as a minimum, current operating history, problems and changes to procedure and administrative limitations. In the event an instructor is a licensed SRO, his retraining shall be the SRO requalification program.

Clarification

The above position is a short-term position. In the future, accreditation of training institutions will include review of the procedure for certification of instructors. The certification of instructors may, or may not, include successful completion of an SRO examination.

The purpose of the examination is to provide NRC with reasonable assurance during the interim period, that instructors are technically competent.

The requirement is directed to permanent members of training staff who teach the subjects listed above, including members of other organizations who routinely conduct training at the facility. There is no intention to require guest lecturers who are experts in particular subjects (reactor theory, instrumentation, thermodynamics, health physics, chemistry, etc.) to successfully complete an SRO examination. Nor is it intended to require a system expert, such as the instrument and control supervisor teaching the control rod drive system, to sit for an SRO examination.

Discussion and Conclusions

The applicant has committed that, prior to fuel loading, all permanent members of the station staff who teach the topics outlined above, will be required to successfully complete an SRO examination and later go through appropriate retraining or requalification programs. Applicant has also committed that contract training personnel, including training center and simulator instructors, will be certified by their management, as to their SRO qualifications. Only those who pass the NRC examinations will continue teaching. The use of guest lecturers who are not required to take SRO examiations will be limited by the applicant. Based on the foregoing, we have concluded that The Cincinati Gas and Electric Company has complied with the requirements of this task of the action plan.

The Office of Inspection and Enforcement will verify that all permanent members of the station staff who teach the topics outlined above have completed an SRO examination prior to fuel loading.

I.A.3.1 Revise Scope and Criteria for Licensing Examinations

Position

Applicants for operator licenses will be required to grant permission to the NRC to inform their facility management regarding the results of examinations.

Contents of the licensed operator requalification program shall be modified to include instruction in heat tranfer fluid flow, thermodynamics, and mitigation of accidents involving a degraded core.

The criteria for requiring a licensed individual to participate in accelerated requalification shall be modified to be consistent with the new passing grade for issuance of a license.

Requalification programs shall be modified to require specific reactivity control manipulations. Normal control manipulations, such as plant or reactor startups, must be performed. Control manipulations during abnormal or emergency operation shall be walked through and evaluated by a member of the training staff. An appropriate simulator may be used to satisfy the requirements for control manipulations.

Clarification

The clarification does not alter the staff's position regarding simulator examinations.

The clarification does provide additional preparation time for utility companies and NRC to meet examination requirements as stated. A study is under way to consider how similar a nonidentical simulator should be for a valid examination. In addition, present simulators are fully booked months in advance.

Application of this requirement was stated on June 1, 1980 to applicants where a simulator is located at the facility. Starting October 1, 1981, simulator examinations will be conducted for applicants of facilities that do not have simulators at the site.

NRC simulator examinations normally require 2 to 3 hours. Normally, two applicants are examined during this time period by two examiners.

Utility companies should make the necessary arrangements with an appropriate simulator training center to provide time for these examinations. Preferably these examinations should be scheduled consecutively with the balance of the examination. However, they may be scheduled no sooner than 2 weeks prior to and no later than 2 weeks after the balance of the examination.

Discussion and Conclusions

The new subject matter and exam grading criteria will be implemented by the NRC with the first license exams given at Zimmer.

The requirement for applicants for operator licenses to grant permission to the NRC to inform facility management regarding results of examinations will also be implemented with the first license examination at Zimmer. The requalification program contents, passing criteria, and control manipulation requirements will be implemented by the applicant upon initiation of the requalification program at Zimmer.

The applicant will make appropriate arrangements with a simulator training center to provide time for license applicant simulator exams. The applicant will make every effort to schedule these examinations consecutively with the balance of the exams. However, due to the constraints of the licensing schedule and training center schedules, the applicant may request a variation in schedule for simulator exams consecutively with the balance of the exams. If it becomes necessary to deviate from the recommended schedule, the applicant will make formal a request for the change.

Based on the foregoing we have concluded that The Cincinati Gas and Electric Company has complied with the requirements of this task of the action plan.

I.B Support Personnel

I.B.1.2 Organization and Management

Position

Corporate management of the utility-owner of a nuclear power plant shall be sufficiently involved in the operational phase activities, including plant modifications, to assure a continual understanding of plant conditions and safety considerations. Corporate management shall establish safety standards for the operation and maintenance of the nuclear power plant. To these ends, each utility-owner shall establish an organization, parts of which shall be located onsite, to: perform independent review and audits of plant activities; provide technical support to the plant staff for maintenance, modifications, operational problems, and operational analysis; and aid in the establishment of programmatic requirements for plant activities.

The licensee shall establish an integrated organizational arrangement to provide for the overall management of nuclear power plant operations. This organization shall provide for clear management control and effective lines of authority and communication between the organizational units involved in the management, technical support, and operation of the nuclear unit. The key characteristics of a typical organization arrangement are:

1. Integration of all necessary functional responsibilities under a single responsible head.

2. The assignment of responsibility for the safe operation of the nuclear power plant(s) to an upper-level executive position.

Utility management shall establish a group, independent of the plant staff, but assigned onsite, to perform independent reviews of plant operational activities. The main functions of this group will be to evaluate the technical adequacy of all procedures and changes important to the safe operation of the facility and to provide continuing evaluation and assessment of the plant's operating experience and performance.

Discussion and Conclusion

In conjunction with our review of the management structure and technical resources for operation of the Zimmer Station, an NRC review team visited the applicant's corporate offices and the Zimmer Station on March 17-19, 1981. The team was composed of personnel from the Offices of Nuclear Reactor Regulation and Inspection and Enforcement. The review team held group discussions with the applicant's corporate level managers and interviewed both individual corporate level and plant managers, technical support staff and operations and maintenance staff to discuss the applicant's perspective of what the organization is and how it works.

The review team met initially with a group of the applicant's corporate office managers of various support activities related to the Zimmer Station. The review team discussed with the applicant overall organization and resources related to the operation and support for the operation of the Zimmer Station. It also discussed the applicant's responses to the TMI Action Plan Items concerned with organization, technical resources, and administrative procedures.

The applicant informed the review team that it was in the process of reorganizing its corporate management structure to place all of the operations and most of the engineering support activities related to the Zimmer Station under a single Vice President, Nuclear Operations who reports to E. A. Borgmann, the Senior Vice President, Engineering Services and Electric Production. As shown in Figure I.B.1.2-1, reporting to the Vice President, Nuclear Operations in this new management structure are four corporate managers: The Manager, W. H. Zimmer Station; the Manager, Nuclear Engineering; the Manager, Quality Assurance; and the Manager Nuclear Services.

Prior to the reorganization as discussed above, the corporate engineering and quality assurance support for the Zimmer Station was provided by corporate functional organizations that were not dedicated solely to the Zimmer Station. They also provided support service to the applicant's fossil fuel plants. The new Nuclear Operations organization will be dedicated solely to the support of nuclear station activities.

Technical support for licensing, environmental and emergency planning activities for the Zimmer Station are provided and managed by the corporate Licensing and Environmental Affairs organization. Construction activities related to the Zimmer Station are managed by the corporate General Construction organization. Both Licensing and Environmental Affairs and General Construction provide services to fossil stations as well as to the Zimmer Station and they both report directly to the Senior Vice President, Engineering Services and Electric Production.

The applicant plans to reassign as many as is practical of the existing corporate support staff that have been working principally in support of the Zimmer Station to the new Nuclear Operations Organization. It also plans to hire additional engineers to substantially increase the size of this corporate staff that is dedicated to the technical support of the Zimmer Station. The applicant has not filled the new Vice President, Nuclear Operations position at this time but has appointed the four managers that report to the Vice President, Nuclear Operations.

Figure I.B.1.2-1

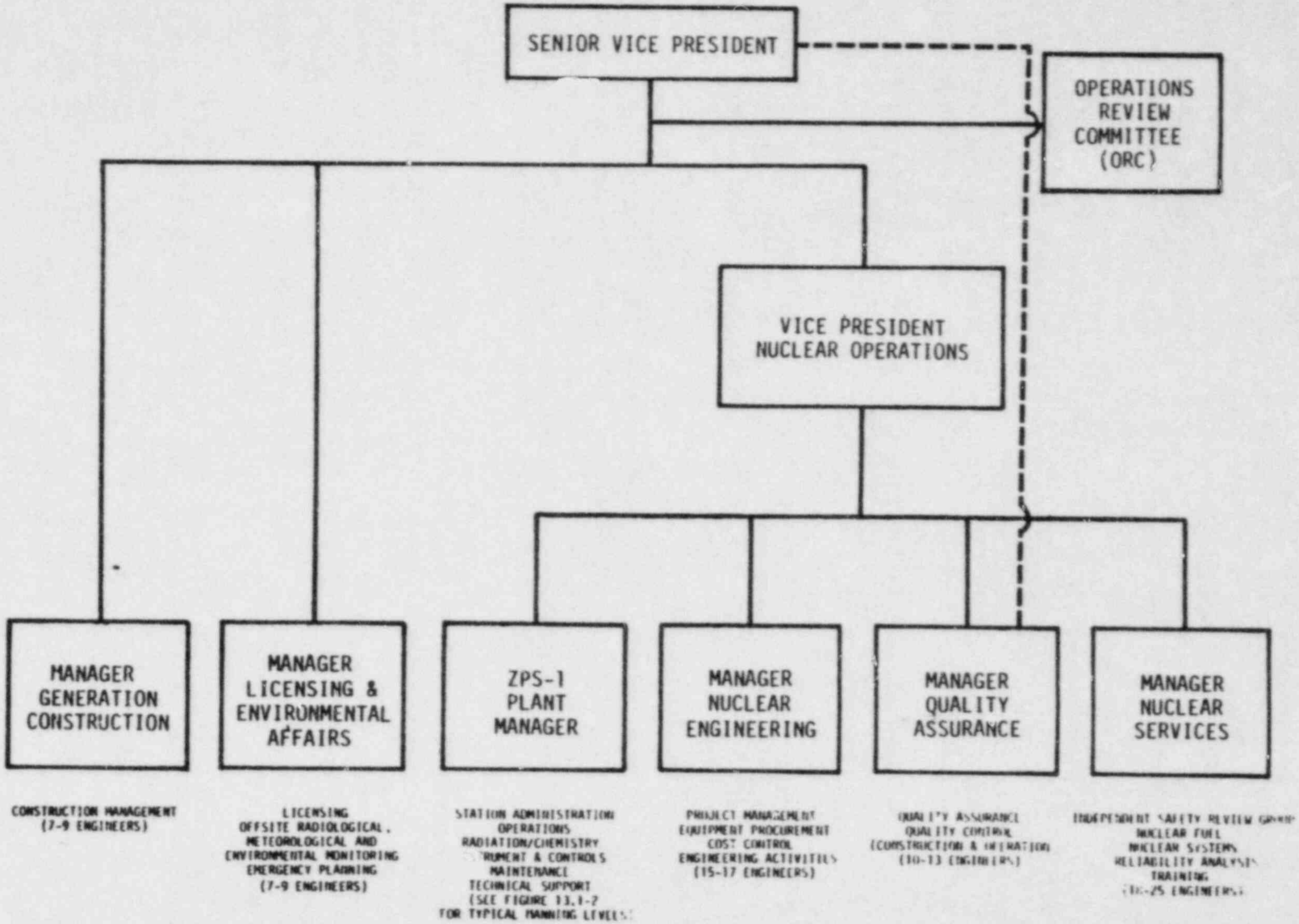

FIGURE I.B.1.2-1

Under the new organization the Zimmer Station Superintendent has been redesignated as the Manager, W. H. Zimmer Station. The organization at the site reporting to this Manager remains essentially as it was before except that the onsite quality assurance staff and the training coordinator who previously reported to the Station Superintendent will now report offsite to the new Manager, Quality Assurance and the new Manager, Nuclear Services, respectively.

The applicant has formed an Independent Safety Review Group (ISRG) in response to this TMI Action Plan Item requirement for an onsite independent safety engineering group. The ISRG is composed of five engineers, including a group leader. It is located onsite at the Zimmer Station and reports offsite to the Manager, Nuclear Services. The ISRG's duties will include (1) safety reviews and evaluation of operations, security, and quality assurance, and (2) review of LERs. The applicant plans that personnel selected as members of the ISRG will serve at least 1 year in this assignment.

The new Vice President, Nuclear Operations and each of the four managers reporting to him will serve as a member of the Offsite Review Committee (ORC), the applicant's independent review body. As discussed in Section 13 of this supplement, three additional corporate office managers and two outside consultants also serve as members of the ORC. The Senior Vice President, Engineering Services and Electric Production is the chairman of the ORC.

These organizational changes, along with additional details concerning the functions of the revised organizations, have been provided by the applicant in Revision 72 of the FSAR.

The review team also interviewed, individually, each of the following managers and staff personnel from the corporate office and the Zimmer Station:

Corporate Office - Senior Vice President, Engineering Services and Electric Production; Manager, Nuclear Engineering; Manager, Quality Assurance; Manager, Nuclear Services; Group Leader, Independent Safety Review Group; Sponsor Engineer (mechanical).

Zimmer Station - Manager, Zimmer Station; Technical Engineer; Rad Chem Engineer; Maintenance Engineer; Instrumentation and Control Engineer; Training Coordinator; a Shift Supervisor; a Shift Technical Advisor; a Senior Reactor Operator; a Reactor Operator; a Nuclear Plant Operator (Auxilairy Operator); a Rad Chem Technician; an Instrumentation and Control Technician; a Maintenance Foreman; a Senior Maintenance Technician.

The review team questioned the interviewees concerning their individual professional qualifications (e.g., education, work experience and training) related to their current position in the applicant's organization. It also questioned them concerning their individual perspective of: their assigned responsibilities, how they interface with and their communication channels with other corporate or plant staff, the training programs for and methods used for training related to their assignments, and how information concerning operating experience at Zimmer and other plants is disseminated to members of the operating and technical support staff that need it.

One of the important subjects discussed with the applicant was operator training and experience. Based on the group discussions, the review team interviews, and the applicant's submittals to date concerning the actual operating experience of the proposed candidates of Zimmer operating licenses, we do not believe that the operating shift crews have sufficient experience to undertake the initial startup and operation of the Zimmer plant. We informed the applicant that at least one person on each operating shift crew should have at least six months experience at an operating boiling water reactor including experience in startup, transient operation and shutdowns of the type that might be expected at the Zimmer Station during its initial startup and escalation to full power.

The applicant informed us of its ongoing program for sending reactor operators to observe operating plants for 2 weeks at a time during various operating phases in order to accumulate the necessary experience. In its April 22, 1981 submittal, the applicant stated that its shift supervisors have accumulated "on the average," 3 months of experience from this program. However, in our interviews with operators who have participated in these observation trips, some of them indicated that they did not get much operating experience at some of the visits because nothing of consequence was happening at the plant at the time of their visit. This appears to be something that would have to be carefully considered in evaluating the credit that might be given for such training. Notwithstanding such problems, we believe that such training where operators observe startup and transient operation of a nuclear plant similar to the one they will work on are useful and could be structured to provide sufficient in-house "operating experience" as discussed above.

The applicant has subsequently informed us in an April 22, 1981 submittal it will provide an advisor on each shift who has at least 6 months of meaningful operating experience during the startup phase of a boiling water reactor and that it will provide this advisor on each shift from initial fuel load until full power operation or until its own operating personnel have obtained 6 months of meaningful experience, whichever is sooner.

It is our understanding, based on discussions with the applicant, that this experienced person on each shift will have had experience operating boiling water reactors under the transient conditions similar to those that would be expected to occur during the initial startup and through full power operation of the Zimmer Station. On this basis, we conclude that the applicant's proposal will substantially enhance the safe startup of the Zimmer Station and is acceptable.

Another of the important subjects that we discussed with the applicant was the training provided for Zimmer Station personnel. We learned that there are substantial differences in the extent of the training and the manner in which the training is provided for technicians in the maintenance, instrumentation and control, and rad-chem areas. The instrument and control technicians appeared to receive both less structured and less extensive training than did technicians in the other two areas. Also, there appeared to be very little in-house training capability in-place for training and retraining operators. We learned, however, that the applicant has recognized that it had a weakness in this training area and has recently hired a new training supervisor and was in the process of hiring approximately six additional training staff personnel. In addition, as discussed above, the applicant has reassigned the management responsibility for the training function to the corporate Manager, Nuclear Services. The training coordinator who previously reported directly to the assistant plant superintendent will now report directly to the Manager, Nuclear Services. This provides greater visibility to and management focus on training. Although this appears to be a desirable change, the applicant should continue to assess this arrangement to assure that the attention to training activities under the Manager, Nuclear Services are not diluted by the Manager's other responsibilities.

The Rad/Chem Supervisor at Zimmer has a direct line of communication to the Station Superintendent in matters of health and safety that could affect onsite and/or offsite personnel. He reports at the same level as the Operating Supervisor. These items are in agreement with the criteria of NUREG-0731 and Regulatory Guide 8.8. The Station Chemist will act as the backup to the Rad/Chem Supervisor in his absence at Zimmer 1. The December 1979 revision of ANSI 3.1 specifies that individuals temporarily filling the RPM position should have a B.S. degree in science or engineering, 2 years' experience in radiation protection, 1 year of which should be nuclear plant experience, 6 months of which should be onsite. The Station Chemist for Zimmer 1 satisfies these requirements.

NUREG-0731 and NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," specify that there shall be a radiation protection technician onsite at all times when there is fuel in the reactor. This individual is in addition to a radiochemistry technician responsible for radwaste operations and other individuals assigned other than radiation protection duties normally, but who might be qualified in radiation protection procedures. Zimmer will have a rad/chem technician onsite at all times, who will be responsible for both radiation protection and chemistry functions. In addition, they will provide an additional person on backshifts whose function is to operate the liquid and solid radwaste facilities.

Zimmer has a combined chemistry and health physics department. Although NUREG-0731 suggests that the health physics and chemistry departments be separate, Cincinnati Gas and Electric maintains that a combined health physics and chemistry department will function effectively at the Zimmer plant. Zimmer provides a comprehensive training and qualification program for their rad/chem technicians. This program consists of approximately 116 hours of classroom training in chemistry and health physics for trainees/ junior techs and approximately 480 hours of classroom training for techs/ senior techs. This classroom training will be supplemented by on-the-job rad/chem technician training. Senior technician candidates are sent to an operating BWR for a nominal 6 weeks on-the-job Radiation Protection technician training. Technicians in training will be given written examinations at approximately 40 training hour intervals to test their comprehension of topics covered. Those students receiving test scores below 75 percent will be required to repeat the training. Qualification cards will be maintained for all technicians to record their training progress. Technicians will be tested orally or by written examination (1) prior to promotion to the next higher job classification, (2) upon completion of one or more sections procedures within a classification, and (3) at least annually to evaluate their understanding of the training material. Zimmer will conduct continuing training in changes in procedures and will provide formal retraining for all rad/chem technicians on at least an annual basis. This formal retraining will be concentrated in subject areas where the technicians have shown areas of weakness. Based on the high caliber of the rad/chem technicians at Zimmer, and on Zimmer's comprehensive training program, we conclude that the joint Chemistry/Radiation Protection department at Zimmer can satisfactorily perform both the health physics and chemistry functions at the plant.

As a result of our concerns in the area of having a dedicated health phsics technician on backshifts at Zimmer, the applicant has committed to providing a rad/chem technician on backshifts whose duties will include health physics and chemistry only. Radwaste operations will be performed by a separate individual. Based on the resolution of our concern in this area, we find Zimmer's Health Physics Organization acceptable.

Based on our interviews and discussions with members of the applicant's operation and technical support staff and our review of the applicant's submittals as discussed above, we conclude that:

(1) The applicant has made adequate arrangements to assure that personnel with substantial previous BWR operating experience are available on each operating shift.

(2) The applicant recognizes the need to increase and enhance its in-house training capability and has taken steps to do this.

(3) Adequate communications channels exist between plant management and the corporate office.

(4) Corporate and plant technical support staffs and resources should be sufficient to assure appropriate attention is given to normal and emergency operational requirements for the Zimmer Station.

(5) The requirement for the addition of an onsite independent engineering group has been acceptably met.

(6) The applicant's organization and management improvements related to the TMI lessons learned are substantial.

(7) The management structure and technical resources provided for operation and support of the Zimmer Station meet the requirements of this TMI Action Plan Item and are acceptable.

I.C Operating Procedures

I.C.1 Guidance for the Evaluation and Development of Procedures for Transients and Accidents

Position

In our letters of September 13 and 27, October 10 and 30, and November 9, 1979, we required licensees of operating plants, applicants for operating licenses, and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, and to conduct operator retraining (see also Item I.A.2.1 of this report). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980, and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions were included in NUREG-0737, Item I.C.1.

Pending staff approval of the revised analysis and guidelines, the staff will continue the pilot monitoring of emergency procedures described in Item I.C.8 (NUREG-0660). The adequacy of the Boiling Water Reactor Owner's Group guidelines will be identified to each near-term operating licensee during the emergency procedure review.

Discussion and Conclusions

In a submittal dated June 30, 1980, the BWR Owners' Group provided a draft of the generic guidelines for Boiling Water Reactors. The guidelines were developed to comply with Task Action Plan Item I.C.1(3) as clarified by NUREG-0737 and incorporated the requirements for short-term reanalysis of small-break loss-of-coolant accidents and inadequate core cooling (Task Action Plan Items I.C.1(1) and I.C.1(2)). In a letter dated October 21, 1980, from D. G. Eisenhut to S. T. Rogers, the staff indicated that the generic guidelines prepared by General Electric and the BWR Owners' Group were acceptable for trial implementation at the Zimmer Station. Additional information was requested by the staff and was submitted by the Owners' Group on January 31, 1981. This additional information is still under review prior to the staff making a final conclusion on the acceptability of the guidelines for implementation on all Boiling Water Reactors. The guidelines are still considered acceptable for trial implementation at the Wm. H. Zimmer Nuclear Power Station. Based on our review of the emergency procedures developed from the BWR Owners' Group Guidelines, and our observation of the procedures being implemented on a simulator and in a walk-through in the control room, we conclude that the guidelines have been adequately incorporated. This fulfills the requirements of Section I.C.1 of NUREG-0694.

I.C.2 Shift Relief and Turnover Procedures

Position

The licensee shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming of offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console. What to check and criteria for acceptable status shall be included on the checklist.

   c. Identification systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement. (This shall be recorded as a separate entry on the checklist.)

2. Checklists or logs shall be provided for completion by the offgoing and oncoming auxiliary operators and technicians. Such checklists or logs

shall include any equipment under maintenance or test that by itself could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedures (for example, periodic independent verification of system alignments).

Discussion and Conclusions

Shift relief and turnover procedures are provided in Section 5.5 of the Zimmer Station Administrative Directive OS.SAD.01. We reviewed Revision 7 to this administrative directive in conjunction with our March 17-20, 1981 site visit. The directive requires the use of a checklist for shift turnover by all shift crew members. The directive requires the checking and assuring acceptability of critical plant parameters, assurance of availability and proper alignment of systems and identification of systems and components that are in a degraded mode, and comparing their time in degraded mode with the Technical Specification limits in accordance with the requirements of Items 1a, 1b and 1c of the above NRC position. Directive OS.SAD.01 also specifies that a checklist will be provided for completion by offgoing and oncoming non-licensed plant operators in accordance with Item 2 of the above NRC position. We also reviewed instructions concerning shift turnover checklists that were provided by the applicant in its internal memorandum from the Operating Engineer to the Operating Group, OPMEMO 81-25, Revision 00, dated January 16, 1981. This memorandum provides the shift turnover checklist. It also discusses the requirement that the shift supervisor, the operator at the controls, the assistant control operator and the plant operator sign the checklists. It does not, however, include the criteria for acceptable status as required by Item 1b above.

We found that the administrative directive did not:

1. Specifically require the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign the checklist as required by Item 1 above of the staff position. (The OPMEMO discusses signing the checklist by some operators but neglects to list the senior control operator (control room SRO).)

2. Address the requirement of Item 1b above of the staff position that the checklist shall include the criteria for acceptable status.

We have discussed these deficiencies with the applicant and it has agreed to modify its administrative directives to correct them.

The applicant has described its system for evaluating the effectiveness of shift turnover, as required by Item 3 of the above NRC position, in its April 22, 1981 submittal.

Subject to modification as discussed above, we conclude that the applicant's shift turnover procedures meet the requirements of Action Item I.C.2 and are acceptable. The NRC Office of Inspection and Enforcement will review and assure that the station administrative directives are corrected as stated above and implemented prior to fuel load.

I.C.3 Shift Supervisor Responsibilities

This item is included with Section I.A.1.2, Shift Supervisor Duties.

I.C.4 Control Room Access

Position

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., operations supervisor, shift supervisor, and control room operators), to technical advisors who may be requested or required to support the operation, and the predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access.

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside the control room.

Discussion and Conclusions

Control room access procedures are provided by Section 5.6 of the Zimmer Station Administrative Directive OS.SAD.01. We reviewed Revision 7 of this administrative directive in conjunction with our March 17-20, 1981 site visit. The directive establishes the shift supervisor as the person responsible for and having authority to control and limit access of individuals to the control room as required by Item 1 of the above NRC position. Section 5.6 of this administrative directive establishes the line of authority and responsibility in the control room during an emergency, in accordance with Item 2 of the above NRC position, by requiring that all control room activities that might affect operations be authorized by the shift supervisor and a nuclear control operator (licensed operator stationed inside the control room). Section 4.6 of this administrative directive, in conformance with Item 2, delineates a line of succession of individuals licensed as Senior Reactor Operators as the persons in charge of the control room.

However, this administrative directive does not clearly define the line of communication and authority of plant management individuals that are not in direct command of operations as required by Item 2. The applicant has agreed to modify the administrative directives to correct this deficiency.

Subject to the incorporation of a clear definition of the lines of communication and authority, during an emergency, of plant management not in direct command of operation, we conclude that the applicant's administrative directives have acceptably satisfied the requirements of Task Action Item I.C.4. The NRC Office of Inspection and Enforcement will review the Zimmer Administrative Directives to assure that they are modified as discussed above and implemented prior to fuel load.

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

Position

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Clarification

This clarification was provided in NUREG-0737 dated November 1980.

Each utility shall carry out an operating experience assessment function that will involve utility personnel having collective competence in all areas important to plant safety. In connection with this assessment function, it is important that procedures exist to assure that important information on operating experience originating both within and outside the organization is continually provided to operators and other personnel and that it is incorporated into plant operating procedures and training and retraining programs.

Those involved in the assessment of operating experience will review information from a variety of sources. These include operating information from the licensee's own plant(s), publications such as IE Bulletins, Circulars, and Notices, and pertinent NRC or industrial assessments of operating experience. In some cases, information may be of sufficient importance that it must be dealt with promptly (through instructions, changes to operating and emergency procedures, issuance of special precautions, etc.) and must be handled in such a manner to assure that operations management personnel would be directly involved in the process. In many other cases, however, important information will become available which should be brought to the attention of operators and other personnel for their general information to assure continued safe plant operation. Since the total volume of information handled by the assessment group may be large, it is important that assurance be provided that high-priority matters are dealt with promptly and that discrimination is used in the feedback of other information so that personnel are not deluged with unimportant and extraneous information to the deteriment of their overall proficiency. It is important, also, that technical reviews be conducted to preclude premature dissemination of conflicting or contradictory information.

Discussion and Conclusion

The applicant stated in its November 26, 1980 submittal concerning this TMI Action Plan Item that it plans to have the Shift Technical Advisors and the Independent Safety Review Group (onsite safety engineering group) review license event reports and manufacturers' notices for impact on the Zimmer Station. However, it has not, to date, described a system for feedback of information on operating experience to plant operators and to other plant support staff either onsite or in the corporate office. Nor has it developed or provided a procedure for assuring that information on operating experience is appropriately disseminated to Zimmer Station operating and support staff. We will require that the applicant submit a detailed description of its system for disseminating operating experience to plant staff for our review. We will also require that it develop appropriate procedures to assure that the feedback system works.

We will report our evaluation of the system in a future supplement to the SER.

I.C.6 Guidance on Procedures for Verifying Correct Performance of Operating Activities

Position

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations, and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

Discussion and Conclusion

In a letter from J. D. Flynn to H. R. Denton dated April 22, 1981, the applicant committed to implement a system for verification of correct performance of operating activities prior to fuel load. The system described is consistent with the clarification in NUREG-0737. The adequacy of the verification system will be determined by the Office of Inspection and Enforcement.

I.C.7 Nuclear Steam Supply System Vendor Review of Procedures

Position

Obtain NSSS vendor review of power-ascension and emergency operating procedures to further verify their adequacy.

This requirement must be met before issuance of a full power license.

Discussion and Conclusion

The NSSS vendor, General Electric Company, is reviewing the power-ascension test procedures and emergency procedures. In a letter from J. D. Flynn to H. R. Denton dated April 22, 1981, the applicant committed to ensure the review is completed successfully prior to the beginning of low power testing. Based on this commitment, we conclude that the requirements of Item I.C.7 have been met. The staff will conduct a review of the applicant's resolution of vendor comments to confirm the acceptability of the vendor review and its implementation. OIE will audit implementation during routine inspections.

I.C.8 Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants

Position

Correct emergency procedures as necessary based on the NRC audit of selected plant emergency operating procedures (e.g., small-break LOCA, loss of feedwater, restart of engineered safety features following a loss of ac power and steam-line break).

This action will be completed prior to issuance of a full power license.

Discussion and Conclusion

During our review of emergency procedures, we discussed with the applicant on March 12, 1981, the Zimmer plant characteristics and the emergency procedures that were based on the generic guidelines prepared by the BWR Owners' Group that are discussed in Section I.C.1 of this SER. The guidelines, which are based on recognition of critical symptoms and restoring and maintaining key plant parameters in predetermined ranges, are applicable to a wide range of transients and accidents. As a result of these discussions, several minor revisions were made to the procedures. On March 24 and 25, 1981, these revised procedures were employed to respond to simulations of accident and transient conditions. A team of NRC and contractor personnel observed Zimmer operators participating in the simulations of several transients and accidents on the Dresden Simulator. The transients and accidents included loss-of-coolant accidents (LOCA) in a range of break sizes, loss of main feedwater, and recovery from inadequate core cooling. Some transients and accidents were run more than once and equipment failures such as loss of offsite power and failure of one emergency diesel-generator, failure of scram breakers to open (ATWS), and failure of individual components in the emergency core cooling systems were included in the simulated events. Several accident simulations included multiple equipment failures such as loss of all high-pressure safety injection and loss of all (AC) power. During the simulation of the events and following each event, we discussed the operators' actions and the procedures with the operators. As a result of this exercise, some additional changes were made to the draft Emergency Operating Procedures.

On March 26, 1981, the team of NRC and Battelle Pacific Northwest Laboratories personnel observed a team of Zimmer control room operators participate in a walkthrough of the procedures in the Zimmer Control Room. Simulated events included a small-break LOCA, a large break LOCA, and inadequate core cooling. Multiple failures beyong the design basis, such as a draining of the Suppression Pool to the Reactor Building Basement through a structural fault, were simulated in the walkthrough. The procedures were discussed with operations personnel during and after each simulated event. The efficient manner in which the procedures were executed indicated that the emergency procedures were properly sequenced and compatible with the control room equipment and arrangement.

During the final review of the procedures with previously identified changes incorporated, a few changes, that were editorial in nature, were discovered. The applicant has committed to incorporating these final staff comments and to change the procedures from all uppercase text to an acceptable use of uppercase and lowercase text. Based on the review and this commitment, we conclude that the requirements of Section I.C.8 have been met.

Future actions required by additional staff review of a submittal from the BWR Owners' Group dated January 31, 1981, and staff positions developed to implement Task Action Plan Item I.C.9, Long-Term Program for Upgrading of Procedures, may require future revisions to the Emergency Operating Procedures.

I.D.1 Control Room Design Review

Position

Licensees and applicants for operating licenses are required to conduct a detailed control room design review to identify and correct human engineering discrepancies (HEDS). This detailed control room design review is expected to take about a year. Those applicants for operating licenses who are unable to complete this review prior to issuance of a license shall make preliminary design assessments of their control rooms to identify significant HEDS and instrumentation problems and proposed corrective actions and a schedule approved by us for correcting such HEDS. These applicants will be required to complete the more detailed control room design reviews on the same schedule as licensees with operating plants.

Clarification

As a result of these requirements, Cincinnati Gas and Electric Co. (CG&E) performed a PDA of the William H. Zimmer control room and submitted its findings to the NRC.

A Human Factors Engineering Branch (HFEB) team reviewed the CG&E PDA report. After reviewing this assessment, the HFEB team, assisted by human factors consultants from Lawrence Livermore National Laboratory and BioTechnology, Inc., conducted an onsite control room design review audit from February 23 to 27, 1981. All HEDS identified and reported by CG&E in their preliminary assessment were review during the HFEB audit to evaluate the suitability of the proposed corrective actions.

The review team identified a number of HEDS which were documented in a CRDR/Audit report which was transmitted to the applicant. The report categorized the HEDS according to their importance. Observed HEDS were given a priority rating of one, two, or three (high, moderate, low), based on the increased potential for operator error and the possible consequences of that error.

HEDS identified as having a high potential for operator error (Category 1) are required to be corrected before loading fuel. HEDS given a priority rating of 2 must be corrected before operation above 5% power. All HEDS identified with a priority rating of 1 or 2 are presented in Appendix F of this report, along with descriptions of the applicant's commitments to correct these HEDS.

HEDS which were given a priority rating of 3 will be addressed by the applicant in the performance of long-term studies to determine the best and most feasible solutions.

A meeting was held during which identified HEDS were discussed, measures for the correction of most HEDS were resolved, and schedules for correcting HEDS established. In subsequent telephone communications with CG&E, all issues were resolved, and a report containing the applicant's commitments was submitted to NRC.

Discussion and Conclusions

Appendix F to this supplement provides the results of: (a) the applicant's preliminary assessment of the Zimmer control room; (b) the staff's Control Room Design Review (CRDR) conducted onsite, Feburary 23-27, 1981; (c) the staff's draft CRDR report submitted April 1, 1981; (d) a subsequent summary meeting held in Bethesda, April 14, 1981, to discuss the applicant's proposed corrective actions for deficiencies noted in our review/audit report; and (e) the applicant's response to the staff's draft CRDR report, submitted May 1, 1981.

We believe that the improvements as stated within this Appendix F will enhance the operator's detection and response capability, and will lessen the probability of operator error under stressful conditions to permit safe operation of the unit during full power operation. Some deficiencies will be addressed in the Detailed Control Room Design Review/Audit. All other action items identified for improvement will be implemented either prior to loading fuel or prior to increasing power above 5 percent.

We expect that the control room design improvements will be reviewed and verified by the I&E resident inspector or the HFEB.

I.G.1 Training During Low-Power Testing

Position

We require applicants for a new operating license to define and commit to a special low-power testing program approved by NRC to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training.

Clarification

Chapter 14 of the Final Safety Analysis Report describes the applicant's initial test program. The objectives of the initial test program include both training and the acquisition of technical data. This program has been determined by the staff to be acceptable as reported in Section 14 of this report. However, we require the applicant to perform additional testing and training beyond the requirements of the initial test program.

Discussion and Conclusions

To provide BWR applicants with guidance for complying with Item I.G.1 of NUREG-0694, the BWR Owners' Group has prepared guidelines for modifying the initial test programs for BWRs. These guidelines propose intensive integration of plant operators into the conduct of the preoperational and startup test programs, with augmented operator training and repetition of tests as necessary to ensure that all operators get direct experience in performing tests which have important training benefits. The staff has reviewed the guidelines and concluded that use of the guidelines is responsive to the training objectives of I.G.1. In a letter dated March 5, 1981, the applicant committed to use of the guidelines.

To satisfy the objective of a test providing meaningful technical information beyond that obtained in the normal startup test program, the staff position is that the applicant should commit to perform a test which

1. provides meaningful technical information not provided by any of the tests prescribed by Regulatory Guide 1.68, "Initial Test Programs",

2. is similar in scope to the PWR Special Low Power Tests previously determined to be acceptable,

3. poses no undue risk to the health and safety of the public, and

4. poses no undue risk to the plant.

To assist BWR applicants in complying with the above position, the staff proposed a Simulated Loss of Offsite and Onsite Power Test. This test, which is presently being developed, would determine the limitations and capabilities of BWRs to maintain safe reactor and containment conditions in the event of a loss of all AC power (except that AC power which is battery-supplied). In the March 5, 1981 letter, the applicant committed to perform this test.

Based on the applicant's commitments made in its letter of March 5, 1981, the staff concludes that the applicant complies with item I.G.1 requirements for fuel loading. The applicant will submit a detailed test procedure and safety analysis for our review at least four weeks prior to licensing.

II. Siting and Design

II.B.1 Reactor Coolant System Vents

Position

Each applicant shall install reactor coolant system and reactor vessel head high-point vents remotely operable from the control room. The applicant must submit a description of the design, location, size, and poer supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46. In addition, procedures and supporting analysis for operator use of the vents that include the information available to the operator for initiating or terminating vent usage should be submitted. Documentation to meet this item is required by July 1, 1981 and implementation is required by July 1, 1982. Detailed clarification of this requirement is provided in Section II.B.1 of NUREG-0737.

Discussion and Conclusions

This item requires that the applicant make provision for venting noncondensible gases from the reactor coolant system which may inhibit core cooling without significantly increasing the probability of a loss-of-coolant accident or resulting in a challenge to containment integrity.

In a letter dated April 22, 1981 from J. D. Flynn (CG&E) to H. Denton (NRC), the applicant described venting provisions for Zimmer. The primary venting capability is provided by the 13 power-operated safety relief valves. Each of the safety relief valves is seismically and Class 1E qualified and the air supply to the six valves which comprise the automatic depressurization system is seismically qualified. These valves can be manually operated from the control room to vent the reactor coolant system. Emergency procedures undertaken to assure core cooling under accident conditions will at the same time result in system venting and, hence, no specific venting procedures have been provided. Temperature sensors in the valve discharge lines are currently used to indicate valve position. The applicant has indicated that this position indication system will be upgraded in accordance with Item II.D.3.

In addition to the capability provided by the safety relief valves, other reactor coolant system vents have also been included in the original plant design. A reactor coolant vent line located at the top of the reactor vessel is operable from the control room. The elevation of this line permits venting the entire reactor coolant system normally connected to the reactor pressure vessel. Since this vent line is part of the original design for Zimmer, it has already been considered in all the design-basis accident analyses contained in the Final Safety Analysis Report.

A third venting capability is provided by the reactor core isolation cooling system. The reactor core isolation cooling system turbine is driven by main steam drawn from inboard of the main steam isolation valves and exhausting to the suppression pool. The reactor core isolation cooling system is seismically qualified, undergoes periodic testing, provides positive indication of operation in the control room, and is operable from the control room. Emergency procedures undertaken to assure core cooling under accident conditions using reactor core isolation cooling will at the same time result in venting of the reactor coolant system and, therefore, no specific procedures for venting the reactor coolant system using the reactor core isolation cooling system were provided.

The applicant also discussed a post-loss-of-coolant accident condition where noncondensible gases could come out of solution while operating the residual heat removal system in the steam condensing mode. These gases would accumulate at the top of the residual heat removal heat exchanger since this is a system high point and an area of relatively low flow. Gases trapped here can be vented through a 3/4-inch vent line with two Class 1E motor-operated valves operated from the control room. As this vent line and associated valves are part of the original design, they have also been considered in the design-basis accident analysis contained in the Final Safety Analysis Report. To accommodate the continuous release of noncondensibles from the residual heat removal heat exchanger when employed in the steam-condensing mode of long-term cooling, these remote vent valves on the heat exchanger vent line are opened to discharge through a submerged line into the suppression pool.

The applicant provided no additional loss-of-coolant accident analyses because no equipment modifications have been made to demonstrate venting capability, i.e., all equipment discussed is part of the original design. The result of a break in the safety relief valve discharge piping, or any of the other pipelines for the systems enumerated above, would be the same as a small steamline break. A more limiting complete steamline break is part of the Zimmer design basis.

The safety relief valve system described by the applicant satisfies the requirement for venting capability imposed by Item II.B.1, provided positive valve position indication is provided in the control room to which the applicant has committed. The other vents (reactor core isolation cooling, vessel head vent, and residual heat removal heat exchanger vents) provide additional capability. We conclude that the Zimmer design meets the vent requirements of Item II.B.1 with the applicant's commitment to provide valve position indication in the control room.

II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Postaccident Operation

Position

With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," and Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors" (i.e., the equivalent of 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and 1 percent of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas of protection of safety equipment by design changes, increased permanent or temporary shielding, or postaccident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Clarification

The purpose of this item is to ensure that licensees examine their plants to determine what actions can be taken over the short term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident. The actions should be taken pending conclusions resulting in the long-term degraded core rulemaking, which may result in a need to consider additional sources.

Any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. For the purposes of this evaluation, vital areas and equipment are not necessarily the same vital areas or equipment defined in 10 CFR Part 73.2 for security purposes. The security center is listed as an area to be considered as potentially vital, since access to this area may be necessary to take action to give access to other areas in the plant.

The control room, technical support center (TSC), sampling station, and sample analysis area must be included among those areas where access is considered vital after an accident. (Refer to Section III.A.1.2 of this report for discussion of the TSC and emergency operations facility.) The evaluation to determine the necessary vital areas should also include, but not be limited to, consideration of the post-loss-of-coolant accident hydrogen control system, containment isolation reset control area, manual emergency core cooling system alignment area (if any), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. Dose rate determinations need not be for these areas if they are determined not to be vital.

As a minimum, necessary modification must be sufficient to provide for vital system operation and for occupancy of the control room, TSC, sampling station, and sample analysis area.

In order to assure that personnel can perform necessary post-accident operations in the vital areas the following guidance is to be used by licensees to evaluate the adequacy of radiation protection to the operators:

1. Source Term

The minimum radioactive source term should be equivalent to the source terms recommended in Regulatory Guides 1.3, 1.4, 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and Standard Review Plan 15.6.5 with appropriate decay times based on plant design (i.e., assuming the radioactive decay that occurs before fission products can be transported to various systems).

a. Liquid-Containing Systems: 100 percent of the core equilibrium noble gas inventory, 50 percent of the core equilibrium halogen inventory, and 1 percent of all others are assumed to be mixed in the reactor coolant and liquids recirculated by residual heat removal, high-pressure coolant injection, and low-pressure coolant injection, or the equivalent of these systems. In determining the source term for recirculated, depressurized cooling water, assuming that the water contains no noble gases.

b. Gas-Containing Systems: 100 percent of the core equilibrium noble gas inventory and 25 percent of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For vapor-containing lines connected to the primary system (e.g., boiling water reactor steam lines), the concentration of radioactivity shall be determined assuming the activity is contained in the vapor space in the primary coolant system.

2. Systems Containing the Source

Systems assumed in your analysis to contain high levels of radioactivity in a postaccident situation should include, but not be limited to, containment, residual heat removal system, safety injection systems, chemical and volume control system, containment spray recirculation system, sample lines, gaseous radwaste systems, and standby gas treatment systems (or equivalent of these systems). If any of these systems or others that could contain high levels of radioactivity were excluded, you should explain why such systems were excluded. Radiation from leakage of systems located outside of containment need not be considered for this analysis. Leakage measurement and reduction is treated under Section III.D.1.1, "Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs and BWRs." Liquid waste systems need not be included in this analysis. Modifications to liquid waste systems will be considered after completion of Section III.D.1.4, "Radwaste System Design Features To Aid in Accident Recovery and Decontamination."

3. Dose Rate Criteria

The design dose rate for personnel in a vital area should be such that the guidelines of Criterion 19 of the General Design Criteria will not be exceeded during the course of the accident. GDC 19 requires that adequate radiation protection be provided such that the dose to personnel should not be in excess of 5 rem whole body, or its equivalent to any part of the body for the duration of the accident. When determining the dose to an operator, care must be taken to determine the necessary occupancy times in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, in order to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines. These doses are design objectives and are not to be used to limit access in the event of an accident.

a. Areas Requiring Continuous Occupancy: F15 mrem/hr (averaged over 30 days). These areas will require full-time occupancy during the course of the accident. The control room and onsite technical support center are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in Standard Review Plan 6.4.

b. Areas Requiring Infrequent Access: Criterion 19 of General Design Criteria. These areas may require access on an irregular basis, not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant radiochemical/chemical analysis laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples of sites where occupancy may be needed often, but not continuously.

4. Radiation Qualification of Safety-Related Equipment

The review of safety-related equipment which may be unduly degraded by radiation during postaccident operation of this equipment relates to equipment inside and outside of the primary containment. Radiation source terms calculated to determine environmental qualification of safety-related equipment consider the following:

a. Loss-of-coolant accident (LOCA) events which completely depressurize the primary system should consider releases of the source term (100 percent noble gases, 50 percent iodines, and 1 percent particulates) to the containment atmosphere.

b. LOCA events in which the primary system may not depressurize should consider the source term (100 percent noble gases, 50 percent iodines, and 1 percent particulates) to remain in the primary coolant. This method is used to determine the qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment. Non-LOCA events both inside and outside of containment should use 10 percent noble gases, 10 percent iodines, and 0 percent particulate as a source term. The following table summarizes these considerations:

Containment	LOCA Source Term (Noble Gas/Iodine/ Particulate)	Non-LOCA High-Energy Line Break Source Term (Noble Gas/Iodine/Particulate)
Outside	Percent (100/50/1) in reactor coolant system	Percent (10/10/0) in reactor coolant system
Inside	Larger of (100/50/1) in containment or (100/50/1) in reactor coolant system	(10/10/0) In reactor coolant system

Discussion and Conclusions

The Zimmer utilized the NRC-prescribed postaccident source terms described in Regulatory Guide 1.3 (as specified in NUREG-0737) to perform their radiation and shielding design review for vital area access. Core inventory and the movement of radioactivity were modeled using the codes RIBD and RACER, respectively. The applicant used the computer codes ISOSHLD and GGG to determine postaccident dose rates. The radiation sources considered in calculating these dose rates included: (1) direct radiation from contained sources in the Reactor and Auxiliary Buildings; (2) direct radiation from airborne and liquid-borne radioactivity in the primary (drywell) and secondary containment (Reactor Building); (3) immersion dose rates from airborne radioactivity due to primary containment and equipment leakage; and (4) radiation from the plant's effluent plume. As specified in NUREG-0737, the applicant has identified the plant systems which may contain high levels of radioactivity and which are required to function as a result of a LOCA. The systems listed are in agreement with those mentioned in NUREG-0737.

The main control room, the technical support center, the operational support center, and the security center are all areas which will require continuous or frequent occupancy following an accident at the Zimmer Power Station. The integrated dose to these areas for the duration of an accident will be less than 5 rem, whole body, as specified in General Design Criteria (GDC) 19. Other vital areas, accessible on an infrequent basis following an accident, are the radiochemical/chemical laboratories, radwaste control room, diesel fuel unloading areas, postaccident sample station, counting room, computer center, and the power supply center. As recommended in NUREG-0737, the applicant has provided a listing of the projected 30-day integrated postaccident doses to individuals for necessary occupancy times in those vital areas requiring infrequent access following an accident. These doses are all within the allowable limits of 10 CFR Part 20. As specified in NUREG-0737, the applicant must submit postaccident dose rate maps for potentially occupied areas and essential access paths between the occupied areas.

The applicant will add a new postaccident sample station, meeting the dose requirements of NUREG-0737, Item II.B.3, paragraph 1, for the postaccident sampling lab, on the 473 foot level of the auxiliary building. This Postaccident Sampling and Analysis System (PASAS) will be designed to enable personnel to promptly obtain and analyze postaccident reactor coolant and containment atmosphere samples. The applicant has estimated that the integrated dose to personnel performing these sampling operations will be an order of magnitude below the 3 rems whole body and 18 3/4 rems to the extremities specified in NUREG-0737. The PASAS System itself will be shielded with lead and steel and will be surrounded by concrete shielding to prevent radiation scattering. The sample lines leading to the PASAS System will be routed through the shielded main steam tunnel. The PASAS System will be equipped with a ventilation system which draws air out of the sampling panels and through a remote HVAC-train.

Status

The applicant has performed a design review (based on the source terms specified in NUREG-0737) of plant shielding and radiation levels for postaccident operations. As a result of this review, the applicant will make shielding and other modifications that are required for them to meet the postaccident shielding requirements of NUREG-0737 for personnel access. Subject to the submittal of the postaccident dose rate maps described above, we find Zimmer's response to Item II.B.2 acceptable for full power operation.

II.B.3 Postaccident Sampling Capability

Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 rem and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," or 1.4 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactor" release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and casiums (which indicate high fuel temperatures) and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

Clarification

The following items are clarifications of requirements identified in NUREG-0578, NUREG-0660, or the September 13, 1979, October 30, 1979, September 5, 1980 and October 31, 1980 clarification letters.

1. The applicant shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

2. The applicant shall establish an onsite radiological and chemical analysis capability to provide, within the 3-hour time frame established above, quantification of the following:

a. Certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and nonvolatile isotopes);

b. Hydrogen levels in the containment atmosphere;

c. Dissolved gases (e.g., hydrogen), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids; and

d. Alternatively, have inline monitoring capabilities to perform all or part of the above analyses.

3. Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system (e.g., the let-down system, reactor water cleanup system) to be placed in operation in order to use the sampling system.

4. Pressurized reactor coolant samples are not required if the applicant can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or hydrogen gas in reactor coolant samples is considered adequate. Measuring the oxygen concentration is recommended, but is not mandatory.

5. The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water, and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions, the applicant shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the applicant shall provide for the analysis to be completed within [illegible] days. The chloride analysis does not have to be done onsite.

6. The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979 letter from D. G. Eisenhut to all licensees).)

7. The analysis of primary coolant samples for boron is required for PWRs. (Note that Revision 2 of Regulatory Guide 1.97 specifies the need for coolant boron analysis capability at BWR plants.)

8. If inline monitoring is used for any sampling and analytical capability specified herein, the applicant shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

9. The applicant's radiological and chemical sample analysis capability shall include provisions to:

   a. Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guides 1.3 or 1.4 and 1.7, "Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident." Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 μCi/g to 10 Ci/g.

   b. Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of ventilation system design which will control the presence of airborne radioactivity.

10. Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

11. In the design of the postaccident sampling and analysis capability, consideration should be given to the following items:

    a. Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the reactor coolant system or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The postaccident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

    b. The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-efficiency particulate air filters.

12. If gas chromatography is used for reactor coolant analysis, special provisions (e.g., pressure relief and purging) shall be provided to prevent high pressure argon from entering the reactor coolant.

13. This requirement applies to all operating reactors and applicants for operating licenses. Installation should take place by January 1, 1982.

14. Operating License Applicants--Provide a description of the implementation of the position and clarification including P&IDs, together with either (a) a summary description of procedures for sample collection, sample transfer or transport, and sample analysis, or (b) copies of procedures for sample collection, sample transfer or transport, and sample analysis, in

accordance with the proposed review schedule but in no case less than 4 months prior to the issuance of an operating license. A postimplementation review will be performed.

Discussion and Conclusions

In a letter dated April 22, 1981, Cincinnati Gas and Electric provided information to satisfy the requirements for Item II.B.3, Postaccident Sampling. In this letter, the applicant committed to install a high radiation sampling system for obtaining reactor coolant and containment atmosphere samples under degraded core accident condition without excessive exposure, by January 1, 1982. The system will be located in the Auxiliary Building with shielded panels for liquid and gas sampling. Provisions will include manual remote sampling control, ventilation air purging, pumps, dilution services and drains. Samples can be transferred to laboratory analysis using a shielded cart. Analysis will include on-line analysis of hydrogen, oxygen, pH and conductivity. Analysis for chloride will be performed by an outside laboratory on a contractual basis. A pump will be included to obtain drywell liquid samples. Air sampling will be provided for the drywell, suppression chamber and reactor building. The applicant should provide the P&ID drawing(s) and a location figure(s), and an estimate of the sampling and analysis time to enable us to complete our review prior to full power operation.

We find that the proposed postaccident sampling and analysis system to be installed and tested by January 1, 1982, can meet the intent of Item II.B.3 requirement, however, we need the P&ID drawings, location figure and combined time estimate for taking reactor coolant and containment atmosphere samples to complete our evaluation. Our completed evaluation will be provided in a supplement to the Safety Evaluation Report.

II.B.4 Degraded Core Training

Position

We require that the applicant develop a program to ensure that all operating personnel are trained in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. They must then implement the training program.

Clarification

Shift technical advisors and operating personnel from the plant manager through the operations chain to the licensed operators shall receive this training. The training program shall include the following topics:

1. Incore Instrumentation

    a. Use of fixed or movable incore detectors to determine extent of core damage and geometry changes.

    b. Use of thermocouples in determining peak temperatures; methods for extended range readings; methods for direct readings at terminal junctions.

2. Excore Nuclear Instrumentation

   a. Use of excore nuclear instrumentation for determination of void formation; void location basis for excore nuclear instrumentation response as a function of core temperatures and density changes.

3. Vital Instrumentation

   a. Instrumentation response in an accident environment; failure sequence (time to failure, method of failure); indication reliability (actual versus indicated level).

   b. Alternative methods for measuring flows, pressures, levels, and temperatures.

      (1) Determination of pressurizer level if all level transmitters fail.

      (2) Determinaation of letdown flow with a clogged filter (low flow).

      (3) Determination of other reactor coolant system parameters if the primary method of measurement has failed.

4. Primary Chemistry

   a. Expected chemistry results with severe core damage; consequences of transferring small quantities of liquid outside containment; importance of using leak-tight systems.

   b. Expected isotopic breakdown for core damage; for clad damage.

   c. Corrosion effects of extended immersion in primary water; time to failure.

5. Radiation Monitoring

   a. Response of process and area monitors to severe damage; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overranged detector); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

   b. Methods of determining dose rate inside containment from measurements taken outside containment.

6. Gas Generation

   a. Methods of hydrogen generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of noncondensibles.

   b. Hydrogen flammability and explosive limit; sources of oxygen in containment or reactor coolant system.

Managers and technicians in the instrumentation and control, health physics, and chemistry departments shall receive training commensurate with their responsibilities.

Discussion and Conclusions

A training program covering the above requirements is being developed by the applicant and will be complete prior to fuel loading. This training will address the upgraded emergency procedures and contingencies presently being developed.

Based on the foregoing, we have concluded that The Cincinnati Gas & Electric Company training and requalification program meets our requirements for training personnel in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The applicant has committed to complete the training of all operating personnel in the use of installed systems to monitor and control accidents in which the core may be severely damaged. This training must be completed before the issuance of full power license.

The Office of Inspection and Enforcement will verify completion of (1) training program prior to fuel loading and (2) training of all operational personnel prior to full power operations.

II.B.7 Analysis of Hydrogen Control

II.B.8 Rulemaking Proceeding on Degraded-Core Accidents

Position

The accident at Three Mile Island, Unit 2 resulted in a severely damaged core accompanied by the generation and release to containment of hydrogen in excess of those limits considered in current regulations. This accident highlighted the difficulties associated with mitigating the consequences of accidents more severe than the current design basis accidents. As a result, the TMI Action Plan (NUREG-0660), item II.B.8, calls for a rulemaking proceeding for consideration of degraded or melted cores in safety reviews. Additionally, the TMI Action Plan, item II.B.7, discusses analysis of hydrogen control and the need for inerting small containments. Both of these requirements for addressing these items are in NUREG-0694.

Clarification

The staff action concerning item II.B.7 was completed with issuance of the Commission papers (SECY-80-107, -80-107A and -80-107B) which address the technical basis for: 1) the staff position on interim hydrogen control requirements (inerting for small containments); and 2) continued operation and licensing of nuclear power plants pending the rulemaking proceeding. With regard to Zimmer, the staff position is that inerting is necessary.

Discussion and Conclusions

The Zimmer applicant has been informed that we are proceeding with the implementation of the recommendations stated in the Commission paper (SECY-80-399) regarding containment inerting. With regard to Zimmer, our position is that inerting is necessary. Presently the draft interim rule has been published for public comments. Upon receipt and evaluation of the comments, the rule will be modified as required and issued for implementation.

In a letter dated March 24, 1981, the applicant stated its intention to inert the primary containment structure for the Zimmer Unit 1. This commitment constitutes a partial fulfillment of the requirements set forth in Items II.B.7 and II.B.8.

The acceptability of the applicant's response to these tasks is contingent upon acceptability of the design description of the system and the approach used to inert. Completion of our evaluation of this matter will be addressed in a supplement to this report.

II.D.1 Performance Testing of Boiling water Reactor and Pressurized Water Reactor Relief and Safety Valves

Position

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

Clarification

Licensees and applicants shall determine the expected valve operating conditions through the use of analyses of accidents and anticipated operational occurrences referenced in Regulatory Guide 1.70, Revision 2. The single failures applied to these analyses shall be chosen so that the dynamic forces on the safety and relief valves are maximized. Test pressures shall be the highest predicted by conventional safety analysis procedures. Reactor coolant system relief and safety valve qualification shall include qualification of associated control circuitry, piping, and supports, as well as the valves themselves.

1. Performance Testing of Relief and Safety Valves - The following information must be provided in report form by October 1, 1981:

   a. Evidence supported by test of safety and relief valve functionability for expected operating and accident (nonanticipated transients without scram (ATWS)) conditions must be provided to NRC. The testing should demonstrate that the valves will open and reclose under the expected flow conditions.

   b. Since it is not planned to test all valves on all plants, each licensee must submit to NRC a correlation or other evidence to substantiate that the valves tested in the EPRI (Electric Power Research Institute) or other generic test program demonstrate the functionability of as-installed primary relief and safety valves. This correlation must show that the test conditions used are equivalent to expected operating and accident conditions as prescribed in the Final Safety Analysis Report. The effect of as-built relief and safety valve discharge piping on valve operability must also be accounted for, if it is different from the generic test loop piping.

c. Test data including criteria for success and failure of valves tested must be provided for NRC staff review and evaluation. These test data should include data that would permit plant-specific evaluation of discharge piping and supports that are not directly tested.

2. Qualification of PWR Block Valves--Although not specifically listed as a short-term lessons-learned requirement in NUREG-0578, qualification of PWR block valves is required by the NRC Task Action Plan NUREG-0660 under task Item II.D.1. It is the understanding of the NRC that testing of several commonly used block valve designs is already included in the generic EPRI PWR safety and relief valve testing program to be completed by July 1, 1981. By means of this letter, NRC is establishing July 1, 1982 as the date for verification of block valve functionability. By July 1, 1982, each PWR licensee, for plants so equipped, should provide evidence supported by test that the block or isolation valves between the pressurizer and each power-operated relief valve can be operated, closed, and opened for all fluid conditions expected under operating and accident conditions.

3. ATWS Testing--Although ATWS testing need not be completed by July 1, 1981, the test facility should be designed to accommodate ATWS conditions of approximately 3200 to 3500 (Service Level C pressure limit) pounds per square inch and 700 degrees Fahrenheit with sufficient capacity to enable testing of relief and safety valves of the size and type used on operating pressurized water reactors.

Discussion and Conclusions

On October 31, 1980, the staff issued, with Commission approval, NUREG-0737, "Clarification of TMI Action Plan Requirements," which is applicable for Operating Reactor Licensees, Applicants for Operating Licenses and Holders of Construction Permits. This NUREG provides more specific guidance on the implementation of TMI Action Plan Items including, for Item II.D.1, the specification of detailed documentation submittal dates.

To date in response to this Action Plan Item, the BWR - TMI Owners' Group, of which the Applicant is a participant, has contracted with the General Electric Co. to develop and implement a generic test program, the results of which will enable BWR utilities to qualify the safety and relief valves and associated discharge piping for their specific plants. Meetings were held on August 27, 1980, October 22, 1980, February 10, 1981 and March 10, 1981 between the NRC staff and representatives of the General Electric Co. and the BWR Owners' Group to discuss the generic valve and piping qualification program developed by G. E. and the analyses that have been performed to define the program test conditions. This information was submitted to NRC by letter dated September 17, 1980 from D. B. Waters to H. Denton. The staff is currently reviewing these analyses results and by letter from D. Eisenhut to D. B. Waters dated February 10, 1981 has provided formal comments on the proposed test program. The BWR Owners' Group has responded to these comments in a letter dated March 31, 1981 from D. B. Waters to the attention of Darrell G. Eisenhut. We are presently reviewing these responses.

The test program that has been proposed provides for qualification of safety/relief and relief valves and associated discharge piping for low-pressure water conditions (i.e., up to 250±20 psig) which are expected during the Alternate hutdown Cooling Mode in which low-pressure pumps inject cold water into the reactor vessel and the water is subsequently vented through the safety/relief or relief valves back to the suppression pool. The position of the Owners' Group is that for all higher pressure, temperature (steam, two-phase, or liquid) conditions, valves and piping on all BWR's have been qualified by tests, analyses or some combination, or that the operating conditions are of such low probability as regards to frequency of occurrence and effects on public health and safety that the valves and piping need not be specifically qualified for them.

In the October 22, 1980 meeting between the staff and the BWR Owners' Group, it was jointly agreed that we would perform a detailed review of the September 17, 1980 BWR Owners' Group submittal, and, in parallel, the BWR Owners' Group would proceed with the implementation of the proposed low-pressure test program with testing to be completed by July 1, 1981 as required by the Action Plan. The NRC staff agreed to perform a detailed review of the September 17, 1980 submittal on as expeditious a basis as available resources would permit. The NRC staff noted that at the completion of the staff review, if the staff was not in agreement with adequacy of the low-pressure test conditions as proposed by the BWR Owners' Group the NRC would require qualification of safety and relief valves and associated piping for higher pressure conditions possibly including subcooled liquid flow at normal reactor operating or slightly higher pressures and temperatures. This position was reaffirmed by us at the February 10, 1981 meeting. The BWR Owners' Group agreed to comply with the staff position on this matter upon completion of the staff review of the September 17, 1980 submittal. If the final staff position is to require valves and piping to be qualified for higher pressure and temperatures than currently proposed by the BWR Owners' Group, the staff agreed to consider a request for a relief from the Action Plan required testing completion date of July 1, 1981 for the higher pressure and temperature tests.

In a letter dated April 22, 1981, Cincinnatii Gas & Electric (CG&E) responded to the requirements of Action Plan Item II.D.1 as clarified by NUREG-0737. In this response, the applicant committed to participating in the BWR Owners' Group program for testing of safety and relief valves. In addition, CG&E is reviewing the BWR program description and scope to insure that it will be applicable to the Zimmer plant specific valves and piping.

In view of the current status of the staff review of the generic BWR Owners' Group program submitted in response to this Action Plan item, the staff concluded that the applicant has committed to the requirements of this item to the extent practicable at this time. It should be noted that the September 17, 1980 submittal from the BWR Owners' Group did refer to documentation dates which are not completely consistent with those required by NUREG-0737. Also the documentation submittal dates in the applicant's April 22, 1981 response to this item are not completely consistent with the NUREG. The staff is continuing the review of both the September 17, 1980 submittal and the responses to NUREG-0737 received from BWR Applicants and Licensees. After we have completed our review of these submittals, we will arrive at a generic resolution regarding NUREG-0737 documentation submittal dates which will be applicable to all operating reactors. We will require CG&E to provide documentation in accordance with this schedule for the Zimmer safety/relief valves and associated discharge piping.

In addition, if on completion of the NRC staff review of the BWR Owners Group program, the staff requires additional information for pressures and temperatures in excess of those provided in the low pressure test program described above, CG&E will be required to participate in development of the information requested with respect to valve operability and system functionability for these temperatures and pressures on a schedule consistent with that agreed to between BWR Owners Group and the NRC staff.

II.D.3 Direct Indication of Relief and Safety Valve Position

Position

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve position detection device or a reliable indication of flow in the discharge pipe.

Clarification

1. The basic requirement is to provide the operator with unambiguous indication of valve position (open or closed) so that appropriate operator actions can be taken.

2. The valve position should be indicated in the control room. An alarm should be provided in conjunction with this indication.

3. The valve position indication may be safety grade. If the position indication is not safety grade, a reliable single-channel direct indication, powered from a vital instrument bus, may be provided if backup methods of determining valve position are available and are discussed in the emergency procedures as an aid to operator diagnosis of an action.

4. The valve position indication should be seismically qualified consistent with the component or system to which it is attached.

5. The position indication should be qualified for its appropriate environment (any transient or accident which would cause the relief or safety valve to lift) and in accordance with Commission Order of May 23, 1980 (CLI-80-21).

6. It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

   a. the use of this information by an operator during both normal and abnormal plant conditions

   b. integration into emergency procedures,

   c. integration into operator training, and

   d. other alarms during emergency and need for prioritization of alarms.

Discussion and Conclusion

Cincinnati Gas and Electric Company has installed direct indication of relief and safety valve position which employs electromechanical devices mounted on top of the valves. Electrical output from these devices is fed to the control room to provide indication and annuciation. A Class 1E power source is used. Backup to this single channel system is provided by temperature sensors in the valve discharge piping and suppression pool.

Although the cabling, circuitry, and mounting panels for the direct valve position indication system meet environmental and seismic qualification requirements, IEEE Standard 323 (1974) and IEEE Standard 344 (1975) respectively, the electromechanical devices have not been demonstrated to meet these standards. The applicant has proposed to meet the seismic and environmental standards at the first refueling outage.

In addition, a human factors analysis which provides an integrated program for monitoring and responding to open relief/safety valves has been developed.

We conclude that the Zimmer design meets the requirements of our position regarding direct indication of relief/safety valve position indication with the exception of those requirements related to environmental and seismic qualification.

It is our position that fully qualified safety/relief valve position indication devices should be installed at Zimmer prior to fuel loading.

II.E System Design

II.E.4.1 Dedicated Hydrogen Penetrations

Position

Plants using external recombiners or purge systems for postaccident combustible gas control of the containment atmosphere should provide containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single-failure requirements of Criteria 54 and 56 of the General Design Criteria and that are sized to satisfy the flow requirements of the recombiner or purge system.

Clarification

1. An acceptable alternative to the dedicated penetration is a combined design that is single-failure proof for containment isolation purposes and single-failure proof for operation of the recombiner or purge system.

2. The dedicated penetration or the combined single-failure proof alternative shall be sized such that the flow requirements for the use of the recombiner or purge system are satisfied. The design shall be based on 10 CFR 50.44 requirements.

3. Components furnished to satisfy this requirement shall be safety grade.

4. Licensees that rely on purge systems as the primary means for controlling combustible gases following a loss-of-coolant accident should be aware of the positions taken in SECY-80-399, "Proposed Interim Amendments to 10 CFR Part 50 Related to Hydrogen Control and Certain Degraded Core Considerations." This proposed rule, published in the Federal Register on October 2, 1980, would require plants that do not now have recombiners to have the capacity to install external recombiners by January 1, 1982. (Installed internal recombiners are an acceptable alternative to the above.)

5. Containment atmosphere dilution (CAD) systems are considered to be purge systems for the purpose of implementing the requirements of this TMI Task Action item.

Discussion and Conclusions

Zimmer will provide a redundant dedicated penetration for the drywell atmosphere, since that is where all of the sources of ignition are located and there is a spare penetration available. A single dedicated penetration will be provided for the material where there is presently no penetration dedicated to the flammability control system. There is one suitable penetration available for this use and, since there are virtually no ignition sources in the wetwell, the single penetration is considered to be sufficient.

The acceptability of the applicant's response to this task is contingent on the requirement that we receive a description of the system indicating that it meets the redundancy and single-failure requirements of the General Design Criteria 54 and 56 of Appendix A to 10 CFR Part 50, and that it is sized to satisfy the flow requirements of the recombiner. Completion of our evaluation will be addressed in a supplement to this report.

II.E.4.2 Containment Isolation Dependability

Position

(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan (SRP) Section 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and non-essential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

(3) All non-essential systems shall be automatically isolated by the containment isolation signal.

(4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

(5) The containment setpoint pressure that initiates containment isolation for non-essential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position (BTP) CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, item II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days. (A copy of the Staff Interim Position is enclosed as Attachment 1.)

(7) Containment purge and vent isolation valves must close on a high radiation signal.

Clarification

1. The reference to SRP 6.2.4 in position 1 is only to the diversity requirements set forth in that document.

2. For postaccident situations, each nonessential penetration (except instrument lines) is required to have two isolation barriers in series that meet the requirements of General Design Criteria 54, 55, 56, and 57 as clarified by SRP, Section 6.2.4. Isolation must be performed automatically (i.e., no credit can be given for operator action). Manual valves must be sealed closed, as defined by SRP, Section 6.2.4, to qualify as an isolation barrier. Each automatic isolation valve in a nonessential penetration must receive the diverse isolation signals.

3. Revision 2 to Regulatory Guide 1.141 will contain guidance on the classification of essential versus nonessential systems and is due to be issued by June 1981. Requirements for operating plants to review their list of essential and nonessential systems will be issued in conjunction with this guide including an appropriate time schedule for completion.

4. Administrative provision to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting position 4.

5. Ganged reopening of containment isolation valves is not acceptable. Reopening of isolation valves must be performed on a valve-by-valve basis, or on a line-by-line basis, provided that electrical independence and other single-failure criteria continue to be satisfied.

6. The containment pressure history during normal operation should be used as a basis for arriving at an appropriate minimum pressure setpoint for initiating containment isolation. The pressure setpoint selected should be far enough above the maximum observed (or expected) pressure inside containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 1 psi above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 1 psi will require detailed justification. Applicants for an operating license and operating plant licensees that have operated less than 1 year should use pressure

history data from similar plants that have operated more than 1 year, if possible, to arrive at a minimum containment setpoint pressure.

7. Sealed-closed purge isolation valves shall be under administrative control to assure that they cannot be inadvertently opened. Administrative control includes mechanical devices to seal or lock the valve closed, or to prevent power from being supplied to the valve operator. Checking the valve position light in the control room is an adequate method for verifying every 14 hours that the purge valves are closed.

Discussion and Conclusions

The following discussion summarizes the applicant's response and our evaluation for each item stated above.

(1) Diversity in parameters

Table II.E.4.2-1 indicates the parameters at Zimmer. Valves which receive two or more of these signals will satisfy the diversity requirement.

We reviewed the applicant's valve actuation signals in the Zimmer Safety Evaluation of January 1979 and found them to be acceptable.

(2) and (3) Essential and nonessential systems.

The applicant has modified Table 6.2.8 of the Final Safety Analysis Report to include essential and nonessential systems. However, additional information is required from the applicant regarding this modification. Completion of our evaluation of this matter will be addressed in a supplement to this report.

(4) Resetting of Containment Isolation Signals

The applicant indicated that its review revealed several cases in which primary containment isolation is removed by the resetting of a containment isolation signal. The applicant stated that the control of these valves will be modified so that each individual valves remains closed when the isolation logic is reset until the control switch is operated to open a particular valve.

(5) Containment Setpoint Pressure

The applicant indicated that the containment setpoint pressure that initiates containment isolation is set to the minimum valve compatible with normal operating conditions. The containment isolation setpoint pressure for Zimmer is 1.69 psig (drywell pressure). The applicant also indicated that under normal operating conditions fluctuations in the atmospheric barometric pressure as well as heat inputs from such sources as pumps can result in containment pressure increases on the order of 1 psi. Consequently, the applicant feels that the isolation setpoint of 1.69 psig provides adequate margin above the maximum expected operating pressure, and that this margin has proved to be a suitable valve to minimize the possibility of spurious containment isolation. At the same time it is felt that such a low valve provides a very sensitive and positive means of detecting and protecting against breaks and leaks in the reactor coolant system.

TABLE II.E.4.2-1 Signals to Initiate Valve Closure

SIGNAL	DESCRIPTION
A	Reactor vessel Low (Level 3) Water Level
B	Reactor vessel Low-(Level 2) Water Level
C	Main Steamline High Radiation
D	Main Steamline High Flow
E	Main Steamline Low Pressure
F	Main Steamline Tunnel Leak Detection (High Temperature or High Δ Temperature)
G	Shutdown Cooling Reactor Pressure High
H	
J	Condenser Vacuum Low
K	Reactor Water Clean High Differential Flow
L	Drywell Pressure High
M	Plant Exhaust Plenum High Radiation
N	RWCU Leak Detection (High Temperature or High Δ Temperature)
P	High Steamline Pressure
Q	Low Dilution Flow or High Leakage Flow
R	RHR/RCIC Combined Steamline High Differential Pressure (High Flow)
S	RHR Equipment Area Leak Detection (High Temperature or High Δ Temperature)
T	RHR Shutdown Cooling Flow High
U	Reactor Vessel Low (Level 1) Water Level
V	RCIC System
	(1) Steam Tunnel Leak Detection (High Temperature or High Temperature)
	(2) RCIC Equipment Area Leak Detection (High Temperature or High Δ Temperature)
	(3) High Steam Flow
	(4) Low Steamline Pressure
	(5) High Turbine Exhaust Pressure
W	High Temperature at Outlet of Cleanup Non-Regenerative Heat Exchanger or Standby Liquid Control System Actuated
X	Close Through Electrical Interlocks with Other Valves or Pump Motors
Z	Refueling Floor Exhaust Radiation - High
RM	Remote Manual From Control Room

We have reviewed the applicant's containment setpoint pressure valve and justification for this valve and find it acceptable.

(6) Purge Valves

The Zimmer Safety Evaluation Report of January 1979 reviewed the design of the containment purge system based upon the criteria specified in BTP CSB 6-4, "Containment Purging During Normal Plant Operation." The purge system satisfied the General Design Criteria 54 and 56 and was found acceptable.

This acceptance is in accordance with the position on purge valve operability criteria as contained in the TMI Action Plan.

(7) Closure of Purge and Vent Valves on High Radiation Signal

The containment purge and vent isolation valves must close on high radiation signal. The applicant indicated that the purge and vent isolation valves receive signal on plant exhaust plenum high radiation (M) and refueling floor exhaust radiation high (Z) in addition to high drywell pressure (L) and low reactor vessel water level (B). These signals meet the intent of the criteria.

II.F.1 Additional Accident-Monitoring Instrumentation

Attachment 1, Noble Gas Effluent Monitor

Position

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near term.

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

1. Noble gas effluent monitors with an upper range capacity of 10^5 μCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

2. Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA)) concentrations to a maximum of 10^5 μCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of 10.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

1. The use of this information by an operator during both normal and abnormal plant conditions;

2. Integration into emergency procedures;

3. Integration into operator training; and

4. Other alarms during emergency and need for prioritization of alarms.

Clarification

NUREG-0578, Section 2.1.8b provided the basic requirements for this item. Letters dated September 27 and November 9, 1979, provided clarification and NUREG-0660, Item II.F.1 provided the action plan for additional accident-monitoring instrumentation by noble gas effluent radiological monitor requirements. Additional clarification was provided by letters dated September 5 and October 31, 1980.

By summary clarification, the following guidelines were established:

1. Applicants shall provide continuous monitoring of high-level postaccident releases of radioactive noble gases from the plant. Gaseous effluent

monitors shall meet the requirements specified in the enclosed Table II.F.1-1. Typical plant effluent pathways to be monitored are also given in the table.

2. The monitors shall be capable of functioning both during and following an accident. System designs shall accommodate a design-basis release and then be capable of following decreasing concentrations of noble gases.

3. Offline monitors are not required for the pressurized water reactor secondary side main stream safety valve and dump valve discharge lines. For this application, externally mounted monitors viewing the main steam line upstream of the valves are acceptable with procedures to correct for the low energy gammas the external monitors would not detect. Isotopic identification is not required. Note: Offline monitors for steamline monitoring application, while acceptable and probably preferable to externally mounted monitors, were not commercially available as of the date of this evaluation.

4. Instrumentation ranges shall overlap to cover the entire range of effluents from normal (ALARA) through accident conditions.

   The design description shall include the following information:

   a. System description, including:

      (1) instrumentation to be used, including range or sensitivity, energy dependence or response, calibration frequency and technique, and vendor's model number, if applicable;

      (2) monitoring locations (or points of sampling), including description of methods used to assure representative measurements and background correction;

      (3) location of instrument readout(s) and method of recording, including description of the method or procedure for transmitting or disseminating the information or data;

      (4) assurance of the capability to obtain readings at least every 15 minutes during and following an accident; and

      (5) the source of power to be used.

   b. Description of procedures or calculational methods to be used for converting instrument readings to release rates per unit time, based on exhaust air flow and considering radionuclide spectrum distribution as a function of time after shutdown.

5. This requirement applies to all operating reactors and applicants for operating license. Implementation must be completed by January 1, 1982.

6. License applicants should have available for review the final design description of the as-built system, including piping and instrument diagrams together with either (1) a description of procedures for system operation and calibration, or (2) copies of procedures for system operation and calibration. Changes to technical specifications will be required. License

applicants will submit the above details in accordance with the proposed review schedule, but in no case less than 4 months prior to the issuance of an operating license. A post-implementation review will be performed.

TABLE II.F.1-1

HIGH-RANGE NOBLE GAS EFFLUENT MONITORS

REQUIREMENT - Capability to detect and measure concentrations of noble fission products in plant gaseous effluents during and following an accident. All potential accident release paths shall be monitored.

PURPOSE - To provide the plant operator and emergency planning agencies with information on plant releases of noble gases during and following an accident.

DESIGN BASIS MAXIMUM RANGE

Design range values may be expressed in Xe-133 equivalent values for monitors employing gamma radiation detectors or in microcuries per cubic centimeter (μCi/cc) of air at standard temperature and pressure (STP) for monitors employing beta radiation detector (Note: 1R/hr @ 1 ft = 6.7 Ci Xe-133 equivalent for point source). Calibrations with a higher energy source are acceptable. The decay of radionuclide noble gases after an accident (i.e., the distribution of noble gas changes) should be taken into account.

10^5 μCi/cc - Undiluted containment exhaust gases (e.g., drywell purge through the standby gas treatement system).

10^4μCi/cc - Diluted containment exhuast gases (e.g., > 10:1 dilution, as with auxiliary building exhaust air).

- BWR reactor building (secondary containment) exhaust air.

10^3μCi/cc - Buildings with systems containing primary coolant or primarily coolant offgases (e.g., BWR turbine buildings).

10^2μCi/cc - Other release points (e.g., radwaste buildings, fuel handling/storage buildings).

REDUNDANCY - Not required; monitoring the final release point of several discharge inputs is acceptable.

SPECIFICATIONS - (None) Sampling design criteria per ANSI N13.1.

POWER SUPPLY - Vital instrument bus or dependable backup power supply to normal AC.

CALIBRATION - Calibrate monitors using gamma detectors to Xe-133 equivalent (1R/hr @ 1 ft = 6.7 Ci XE-133 equivalent for point source). Calibrate monitors using beta detectors to Sr-90 or similar long-lived beta isotope of at least 0.2 MeV.

TABLE II.F.1-1 (Continued)

DISPLAY -	Continuous and recording as equivalent Xe-133 concentrations or µCi/cc of actual noble gases.
QUALIFICATION -	The instruments shall provide sufficiently accurate responses to perform the intended function in the environment to which they will be exposed during accidents.
DESIGN CONSIDERATIONS -	Offline monitoring is acceptable for all ranges of noble gas concentrations.
-	Inline (induct) sensors are acceptable for 10^2 µCi/cc to 10^5 µCi/cc noble gases. For less than 10^2 µCi/cc, offline monitoring is recommended.
-	Upstream filtration (prefiltering to remove radioactive iodines and particulates) is not required; however, design should consider all alternatives with respect to capability to monitor effluents following an accident.
-	For external mounted monitors (e.g., pressurized water reactor main steam line) the thickness of the pipe should be taken into account in accounting for low-energy gamma radiation.

Discussion and Conclusions

In a letter, dated April 21, 1981, Cincinnati Gas and Electric provided information to satisfy the requirements for II.F.1, Attachment 1, noble gas effluent monitoring. Monitors for radioactive effluents currently installed at Zimmer, Unit No. 1, were designed to detect and measure releases associated with normal reactor operations and anticipated operational occurrences. Such monitors are required to operate in radioactivity concentrations approaching the minimum concentration detectable with "state-of-the-art" sample collection and detection methods. These monitors comply with the criteria of Regulatory Guide 1.21 with respect to releases from normal operations and anticipated operational occurrences.

Radioactive gaseous effluent monitors designed to operate under conditions of normal operation and anticipated operational occurrences do not have sufficient dynamic range to function under release conditions associated with certain types of accidents. General Design Criterion 64 of Appendix A to 10 CFR Part 50 requires that effluent discharge paths be monitored for radioactivity that may be released from postulated accidents.

The potential gaseous effluent release point at Zimmer, Unit No. 1, is the plant vent stack. The applicant has committed to install a mid- and high-level noble gas effluent monitor at this point by February 1982. The monitors will be designed to meet the requirements and satisfy the characteristics given in Table II.F.1-1. The monitor will be the off-line type and the system will include new isokinetic probes.

The applicant has stated that the permanent monitoring equipment will be installed and operational by February 1982 (subject to timely delivery of equipment) and therefore prior to fuel loading. For this reason, the applicant does not plan to provide interim monitoring equipment and procedures, which the staff's requirements in NUREG-0737 specify in the event of reactor operation prior to installation of the equipment. The staff finds the licensee's proposal to be acceptable, but reserves the option to require interim monitoring equipment and procedures described in the September 27, 1979, letter from D. Eisenhut to all pending operating license applicants, in the event of delays in the licensee's procurement and installation of the permanent equipment.

We find that the proposed noble gas effluent monitoring system to be installed by February 1982 can meet the intent of the II.F.1 requirement; however, a post-implementation review of the installed system detailed drawings and the procedures for monitoring and calibration will be performed, and an evaluation provided in a supplement to the Safety Evaluation Report.

ATTACHMENT 2, Sampling and Analysis of Plant Effluents

Position

The requirements associated with this recommendation should be considered as advanced implementation of certain requirements to be included in a revision to Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident," which has already been initiated, and in other Regulatory Guides, which will be promulgated in the near term.

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

1. The use of this information by an operator during both normal and abnormal plant conditions;

2. Integration into emergency procedures;

3. Integration into operator training; and

4. Other alarms during emergency and need for prioritization of alarms.

Clarification

NUREG-0578, Section 3.1.8b provided the basic requirements for this item. Letters dated September 27 and November 9, 1979, provided clarification, however, NUREG-0660 inadvertently omitted this requirement on the action plan for additional accident-monitoring instrumentation by sampling and analysis of plant effluents. Additional clarification was provided by letters dated September 5, and October 31, 1980.

By summary clarification, the following guidelines were established:

1. Applicants shall provide continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates to meet the requirements of the enclosed Table II.F.1-3. Applicants shall also provide onsite laboratory capabilities to analyze or measure these samples. This requirement should not be construed to prohibit design and development of radioiodine and particulate monitors to provide online sampling and analysis for the accident condition. If gross gamma radiation measurement techniques are used, then provisions shall be made to minimize noble gas interference.

2. The shielding design basis is given in Table II.F.1-3. The sampling system design shall be such that plant personnel could remove samples, replace sampling media and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the Criterion 19 of the General Design Criteria of 5-rem whole-body exposure and 75 rem to the extremities during the duration of the accident.

3. The design of the systems for the sampling of particulates and iodines should provide for sample nozzle entry velocities which are approximately isokinetic (same velocity) with expected induct or instack air velocities. For accident conditions, sampling may be complicated by a reduction in stack or vent effluent velocities to below design levels, making it necessary to substantially reduce sampler intake flow rates to achieve the isokinetic condition. Reductions in air flow may well be beyond the capability of available sampler flow controllers to maintain isokinetic conditions; therefore, the staff will accept flow control devices which have the capability of maintaining isokinetic conditions with variations in stack or duct design flow velocity of $\pm$ 20 percent. Further departure from the isokinetic condition need not be considered in design. Corrections for anisokinetic sampling conditions, as provided in Appendix C of ANSI 13.1-1969 may be considered on an ad hoc basis.

4. Effluent streams which may contain air with entrained water, e.g., air ejector discharge, shall have provisions to ensure that the adsorber is not degraded while providing a representative sample, e.g., heaters.

5. This requirement applies to all operating reactors and applicants for operating license. Implementation must be completed by January 1, 1982.

6. License applicants should have available for review the final design description of the as-built system, including piping and instrument diagrams together with either (a) a description of procedures for system operation and calibration, or (b) copies of procedures for system operation and calibration. Changes to technical specifications will be required. Applicants will submit the above details in accordance with the proposed review schedule, but in no case less than 4 months prior to the issuance of an operating license. A postimplementation review will be performed.

TABLE II.F.1-2

SAMPLING AND ANALYSIS OR MEASUREMENT OF HIGH RANGE RADIOIODINE AND PARTICULATE EFFLUENTS IN GASEOUS EFFLUENT STREAMS

EQUIPMENT	-	Capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident. The capability to sample and analyze for radioiodine and particulate effluents is not required for pressurized water reactor secondary main steam safety valve and dump valve discharge lines.
PURPOSE	-	To determine quantitative release of radioiodines and particulates for dose calculation and assessment.
DESIGN BASIS SHIELDING ENVELOPE	-	10^2 μCi/cc of gaseous radioiodine and particulates, deposited on sampling media; 30 minutes sampling time, average gamma energy (E) of 0.5 MeV.

SAMPLING MEDIA

- Iodine > 90 percent effective adsorption for all forms of gaseous iodine.
- Particulates > 90 percent effective retention for 0.3 micron (μ) diameter particles.

SAMPLING CONSIDERATIONS

- Representative sampling per ANSI N13.1-1969.
- Entrained moisture in effluent stream should not degrade adsorber.
- Continuous collection required whenever exhaust flow cocurs.
- Provisions for limiting occupational dose to personnel incorporated in sampling systems, in sample handling and transport, and in analysis of samples.

ANALYSIS

- Design of analytical facilities and preparation of analytical procedures shall consider the design basis sample.
- Highly radioactive samples may not be compatible with generally accepted analytical procedures; in such cases, measurement of emissive gamma radiations and the use of shielding and distance factors should be considered in design.

Discussions and Conclusions

In a letter dated April 22, 1981, Cincinnati Gas and Electric provided information to satisfy the requirements for II.F.1, Attachment 2, sampling and analysis of plant effluents.

Monitors for radioactive effluents currently installed at Zimmer, Unit No. 1, are designed to detect and measure releases associated with normal reactor operations and anticipated operational occurrences. Such monitors are required to operate in radioactivity concentrations approaching the minimum concentration detectable with "state-of-the-art" sample collection and detection methods. These monitors comply with the criteria of Regulatory Guide 1.21 with respect to releases from normal operations and anticipated operational occurrences.

The potential gaseous effluent release point at Zimmer, Unit No. 1, is the same as for the noble gases (II.F.1, Attachment 1). The applicant has committed to install new isokinetic probes in the release line by January 1982. The applicant will utilize the new and existing isokinetic probes and the normal off-line sampling system to quantify release rates during an accident.

We find that the radioiodine and particulate sampling and analysis system to be installed by January 1982 can meet the intent of the II.F.1 requirement, however a post-implementation review of the installed system detailed drawings and the procedures for sampling will be performed, and an evaluation provided in a supplement to the Safety Evaluation Report.

ATTACHMENT 3, Containment High-Range Radiation Monitor

Position

In containment radiation-level monitors with a maximum range of 108 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

Clarification

1. Provide two radiation monitor systems in containment which are documented to meet the requirements of Table II.F.1-4.

2. The specification of 108 rad/hr in the above position was based on a calculation of postaccident containment radiation levels that included both particulate (beta) and photon (gamma) radiation. A radiation detector that responds to both beta and gamma radiation cannot be qualified to post-LOCA (loss-of-coolant accident) containment environments but gamma-sensitive instruments can be so qualified. In order to follow the course of an accident, a containment monitor that measures only gamma radiation is adequate. The requirement was revised in the October 30, 1979 letter to provide for a photon-only measurement with an upper range of 107 R/hr.

3. The monitors shall be located in containment(s) in a manner as to provide a reasonable assessment of area radiation conditions inside containment. The monitors shall be widely separated so as to provide independent measurements and shall "view" a large fraction of the containment volume. Monitors

should not be placed in areas which are protected by massive shielding and should be reasonably accessible for replacement, maintenance, or calibration. Placement high in a reactor building dome is not recommended because of potential maintenance difficulties.

4. For BWR Mark III containments, two such monitoring systems should be inside both the primary containment (drywell) and the secondary containment.

5. The monitors are required to respond to gamma photons with energies as low as 60 keV and to provide an essentially flat response for gamma energies between 100 keV and 3 MeV, as specified in Table II.F.1-4. Monitors that use thick shielding to increase the upper range will underestimate postaccident radiation levels in containment by several orders of magnitude because of their insensitivity to low energy gamms and are not acceptable.

Discussion and Conclusions

The applicant will install two, physically separate, gamma-sensitive radiation monitors in the primary containment penetrations. These monitors will be supplied by Victoreen and will meet the specifications in Table II.F.1-3 of NUREG-0737. These two monitors will be located so as to view a large segment of the containment atmosphere and will be capable of monitoring a full range of 10^0 to 10^7 R/hr. These monitors will be calibrated in-situ using electronic signal substitution for all ranges above 10 R/hr. Calibration for lower dose ranges can be performed with calibration sources without disconnecting the detector from the readout module.

The containment high range radiation monitors will be mounted in steel sleeves protruding into the primary containment. NUREG-0737 states that high range containment monitors must be sensitive down to 60 KeV photons. Although the Victoreen monitors used at Zimmer have a range of 60 KeV photons to 3 MeV, the installation in sleeves will result in the attenuation of the lower energy gammas by a factor of approximately four. The applicant has stated that he will use grab sampling in conjunction with onsite isotopic analysis to supplement the high range containment monitor readings and provide dose quantification data for the lower energy photons (including the 81 KeV photons of Xe^{133}). Although, grab sample analysis will provide isotopic information on the presence of Xe^{133} and other low energy gammas, this method may require several hours of analysis time before the desired information is obtained. The applicant must also have a means of determining the low energy gamma contribution to the overall containment radiation levels in a more expeditious manner. Use of pre-established radiation level correlations will allow quick identification of the actual containment radiation levels (including consideration of the 81 KeV photons), and will assure proper operator response to the indicated levels provided by the high range containment radiation monitors. Cincinnati Gas and Electric must implement a method to quickly determine the post-accident dose rate in containment using the dose rate measured by the radiation monitors in the containment penetrations.

Status

The applicant has adequately addressed the criteria of Item II.F.1. Their response meets the positions set forth in NUREG-0737 and is acceptable subject to receipt of a description of methods to be used to correlate high range containment monitor readings with actual in-containment radiation levels.

TABLE II.F.1-3

CONTAINMENT HIGH-RANGE RADIATION MONITOR

REQUIREMENT	-	The capability to detect and measure the radiation level within the reactor containment during and following an accident.
RANGE	-	1 rad/hr to 10^8 rads/hr (beta and gamma) or alternatively 1 R/hr to 10^7 R/hr (gamma only).
RESPONSE	-	60 KeV to 3 MeV photons, with linear energy response ± 20% for photons of 0.1 MeV to 3 MeV. Instruments must be accurate enough to provide usable information.
REDUNDANT	-	A minimum of two physically separated monitors (i.e., monitoring widely separated spaces within containment).
DESIGN AND QUALIFICATION	-	Category 1 instruments as described in Appendix B, except as listed below.
SPECIAL CALIBRATION	-	In-situ calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. In-situ calibration for at least one decade below 10 R/hr shall be by means of calibration is not an acceptable position due to the possible differences after in-situ installation. For high-range calibration, no adequate sources exist, so an alternate was provided.
SPECIAL ENVIRONMENTAL QUALIFICATION	-	Calibrate and type-test respresentative specimens of detectors at sufficient points to demonstrate linearity through all scales up to 10^6 R/hr. Prior to initial use, certify calibration of each detector for at least one point per decade of range between 1 R/hr and 10^3 R/hr.

ATTACHMENT 4, Containment Pressure Monitor

Position

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and -5 psig for all containments.

Clarification

1. Design and qualification criteria are outlined in Appendix B of NUREG-0737.

2. Measurement and indication capability shall extend to 5 pounds per square inch absolute for subatmospheric containments.

3. Two or more instruments may be used to meet the range requirements. However, instruments that need to be switched from one scale to another scale to meet the range requirements are not acceptable.

4. Continuous display and recording of the containment pressure over the specified range in the control room is required.

5. The accuracy and response time specifications of the pressure monitor shall be provided and justified to be adequate for their intended function.

Discussion and Conclusions

Zimmer will have two additional channels for measuring drywell pressure. The instruments will have a range of -10 to 140 psig which is three time the concrete design pressure. One channel will be continuously displayed on an indicating recorder in the control room. The other channel will be supplied to the Technical Support Center and will be recorded in the computer system as well as displayed on demand on a CRT in the control room and the Technical Support Center.

The minimum accuracy of the transmitter/indicator system for the -10 to 45 psig pressure range under normal operating conditions is ± 1.5, while under DBA conditions it is ± 2.5 psig. Beyond this range the minimum accuracy of the transmitters is ± 8 psig. The overall response time of the installed system (pressure transmitter through recorder output) is less than 1 second.

We conclude, therefore, that Zimmer complies with the provisions of Item II.F.1, Attachment 4. (See Section 3.11 of this report for the applicant's Environmental Qualification Program).

ATTACHMENT 5, Containment Water Level Monitor

Position

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for pressurized water reactors (PWRs) and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for boiling water reactors (BWRs) and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

Clarification

1. The containment wide-range water level indication channels shall meet the design and qualification criteria as outlined in Appendices B and C. The narrow-range channel shall meet the requirements of Regulatory Guide 1.89.

2. The measurement capability of 600,000 gallons is based on recent plant designs. For older plants with smaller water capacities, licensees may propose deviations from this requirement based on the available water supply capability at their plant.

3. Narrow-range water level monitors are required for all sizes of sumps but are not required in those plants that do not contain sumps inside the containment.

4. For BWR pressure-suppression containments, the emergency core cooling system (ECCS) suction line inlets may be used as a starting reference point for the narrow-range and wide-range water level monitors, instead of the bottom of the suppression pool.

5. The accuracy requirements of the water level monitors shall be provided and justified to be adequate for their intended function.

Discussion and Conclusions

Zimmer will have two additional channels for measuring suppression pool water level. These instruments will measure water level over a 10-foot 6-inch range from approximately 5 feet above normal water level to 5 feet 6 inches below normal water level. One channel will be continuously displayed on an indicating recorder in the control room. The signal from the other channel will be supplied to the Technical Support Center computer and will be recorded in the computer system as well as displayed on demand on CRT's in the control room and the Technical Support Center. The monitors will meet the minimum accuracy of ±5% of the monitoring range which we judge to be acceptable.

The acceptability of the applicant's response to this task is contingent on our receipt of a commitment to expand the recorded range to the ECCS suction line inlet as the starting reference for the water level monitor. (See Section 3.11 of this report for the applicant's Environmental Qualification Program.)

ATTACHMENT 6, Containment Hydrogen Monitor

Position

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10 percent hydrogen concentration under both positive and negative ambient pressure.

Clarification

1. Design and qualification criteria are outlined in Appendix B.

2. The continuous indication of hydrogen concentration is not required during normal operation.

   If an indication is not available at all times, continuous indication and recording shall be functioning within 30 minutes of the initiation of safety injection.

3. The accuracy and placement of the hydrogen monitors shall be provided and justified to be adequate for their intended function.

Discussion and Conclusions

Zimmer monitors hydrogen concentration using redundant Delphi K-II hydrogen analyzers. The measurement capability of these devices is 0 to 10% hydrogen concentration under ambient pressure conditions of 12 psia to 60 psig. The hydrogen monitors start automatically immediately at the onset of a LOCA. These monitors have an accuracy of $\pm 2\%$ when corrected for temperature and steam conditions within the containment. Hydrogen concentrations and distribution as well as sampling point locations are discussed in the FSAR Question Q41.33 and Figure Q041.33-1. We find the monitoring system as proposed to be acceptable.

We conclude, therefore, that the Zimmer applicant complies with the provisions of Item II.F.1, Attachment 6. (See Section 3.11 of this report for the applicant's Environmental Qualification Program).

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

Position

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Clarification

1. Design of new instrumentation should provide an unambiguous indication of ICC. This may require new measurements or a synthesis of existing measurements which meet design criteria (item 7).

2. The evaluation is to include reactor-water-level indication.

3. Licensees and applicants are required to provide the necessary design analysis to support the proposed final instrumentation system for inadequate core cooling and to evaluate the merits of various instruments to monitor water level and to monitor other parameters indicative of core-cooling conditions.

4. The indication of ICC must be unambiguous in that it should have the following properties:

    a. It must indicate the existence of inadequate core cooling caused by various phenomena (i.e., high-void fraction-pumped flow as well as stagnant boil-off); and

    b. It must not erroneously indicate ICC because of the presence of an unrelated phenomenon.

5. The indication must give advanced warning of the approach of ICC.

6. The indication must cover the full range from normal operation to complete core uncovery. For example, water-level instrumentation may be chosen to provide advanced warning of two-phase level drop to the top of the core and could be supplemented by other indicators such as incore and core-exit thermocouples provided that the indicated temperatures can be correlated

to provide indication of the existence of ICC and to infer the extent of core uncovery. Alternatively, full-range level instrumentation to the bottom of the core may be employed in conjunction with other diverse indicators such as core-exit thermocouples to preclude misinterpretation due to any inherent deficiencies or inaccuracies in the measurement system selected.

7. All instrumentation in the final ICC system must be evaluated for conformance to Appendix B of NUREG-0373, "Clarification of TMI Action Plan Requirements," as clarified or modified by the provisions of items 8 and 9 that follow. This is a new requirement.

8. If a computer is provided to process liquid-level signals for display, seismic qualification is not required for the computer and associated hardware beyond the isolator or input buffer at a location accessible for maintenance following an accident. The single-failure criteria of item 2, Appendix B, need not apply to the channel beyond the isolation device if it is designed to provide 99 percent availability with respect to functional capability for liquid-level display. The display and associated hardware beyond the isolation device need not be Class 1E, but should be energized from a high-reliability power source which is battery backed. The quality assurance provisions cited in Appendix B, item 5, need not apply to this portion of the instrumentation system. This is a new requirement.

9. Incore thermocouples located at the core exit or at discrete axial levels of the ICC monitoring system and which are part of the monitoring system should be evaluated for conformity with Attachment 1, "Design and Qualification Criteria for PWR Incore Thermocouples," which is a new requirement.

10. The types and locations of displays and alarms should be determined by performing a human-factors analysis taking into consideration:

    a. the use of this information by an operator during both normal and abnormal plant conditions,

    b. integration into emergency procedures,

    c. integration into operator training, and

    d. other alarms during emergency and need for prioritization of alarms.

Discussion and Conclusion

Introduction

A clarification of requirements for inadequate core cooling (ICC) instrumentations, which are required to be installed and operational prior to fuel load was provided in H. Denton's letter to All Operating Nuclear Power Plants on "Discussion of Lessons Learned Short-Term Requirements," dated October 30, 1979, and in Section II.F.2 of NUREG-0737 "Clarification of TMI Action Plan Requirements." The staff will use the requirements specified in NUREG-0737 to review applicant's submittals in response to Section II.F.2 of the TMI Action Plan.

Inadequate Core Cooling Detection System

The Cincinnati Gas and Electric Company is a member of the BWR Owner's Group, which has transmitted attachments in a letter (BWR 06-8117 dated January 31, 1981 from D. B. Waters to D. G. Eisenhut (NRC)). Unambiguous procedures to recognize the approach of ICC are given in one of those attachments "BWR Emergency Procedure Guidelines, Revision (January 30, 1981)." These procedures rely primarily on decreasing reactor vessel water level as an indication of the approach to ICC. The water level instruments are part of the standard BWR/5 and later BWR/4 level monitoring systems which were evaluated by the BWR Owners' Group and documented in NEDO-24708. The wide-range instruments when offscale low are presumed by the operator to indicate the approach to ICC. Fuel-zone level measurement is an alternate method for detection of ICC.

Adequate core cooling is assured whenever the reactor is shut down and one or more of the following conditions exist:

(1) the active fuel is covered with liquid or a two-phase mixture;
(2) ECCS flow is cooling each fuel assembly in sufficient quantity to remove all heat generated in the assembly; and
(3) steam flow is cooling each fuel assembly in sufficient quantity to remove all heat generated in the assembly.

One core spray system operating alone provides two means of cooling the core (spray and reflood). The emergency guidelines/procedures specify the minimum water level in the core which is sufficient to remove decay heat by steam cooling alone.

Other than water level, there is no instrumentation available which will provide an unambiguous indication of an approach of ICC. The source range monitors will show changes in water level in the core but they may be misleading as to whether the level is increasing or decreasing. Incore thermocouples, if they were available, might indicate bundle uncovery, but due to spray from the spray systems above the core, might indicate saturated or subcooled conditions even when the core is partially uncovered. However, under certain core uncovery conditions incore thermocouples could indicate coolant superheat associated with excessive fuel cladding temperature.

All events that threaten the ability to provide adequate core cooling have one common factor: the reactor water level decreases. This is true whether the event is the loss of makeup water as in loss-of-feedwater transients, a sustained imbalance between feedwater flow and steam flow as in feedwater controller failure transients, or an excessive loss of liquid or steam inventory as in loss-of-coolant accidents. The principal method of confirming adequate core cooling, namely, the reactor pressure vessel water level instrumentation, has been shown through analysis and experience to be adequate to assure detection of approach to ICC.

Reactor vessel level is measured by differential pressure devices. Condensing chambers connected to the steam space in the reactor vessel furnish saturated water to the reference leg. Pressure taps located at different levels in the water space of the reactor vessel are used as the variable leg sensing taps for the water level instruments.

Because of the presence of voids in the fluid inside the shroud (core area), the swollen level inside the shroud will be higher than the level outside the shroud until ECCS operation commences. Thus, the level measurement in the annular region is a conservative measure of the two-phase level inside the shroud prior to ECCS injection. ECCS injection (which injects directly inside the shroud), will subcool the liquid inside the shroud, leading to the possibility that the level outside the shroud could be somewhat lower than the level outside the shroud. However, since the core is adequately cooled whenever the ECCS is injecting, this discrepancy is of no concern.

Evaluation

Analyses presented by the BWR Owners' Group show the level instrumentation to be adequate for predicating the approach to ICC and for providing a basis for operator action. The analyses are found to be acceptable to support emergency procedure guidelines for ICC.

In a submittal of June 30, 1980, the BWR Owners' Group provided a draft of the generic guidelines for boiling water reactors emergency procedures. The guidelines were developed to comply with Task Action Plan Item I.C.1(3) as clarified by NUREG-0737 and incorporated the requirements for short term reanalysis of small break, loss-of-coolant accidents and ICC (Task Action Plan Items I.C.1(1) and I.C.1(2)). In a letter dated October 21, 1980, from D. G. Eisenhut to S. T. Rogers, the staff indicated that the generic guidelines prepared by GE and the GE Owners' Group were acceptable for trial implementation at the Zimmer-1 plant. Additional information was requested by the staff and was submitted by the Owners' Group on January 31, 1981. This additional information is still under review prior to the staff making a final conclusion on the acceptability of the guidelines for implementation on all boiling water reactors. The guidelines are still considered acceptable for trial implementation at the Zimmer-1 plant. The staff conducted walk-through in a simulator and control room and found that the guidelines have been adequately incorporated.

The applicant has provided the emergency procedures based on guidelines submitted by the owners' group on January 31, 1981. Future actions required by additional staff review of a submittal from the GE Owners' Group committed on January 31, 1981, and staff positions developed to implement Task Action Item I.C.9, Long-Term Program for Upgrading for Procedures, may require future revisions to the emergency procedures.

The GE Owners' Group has concluded that no additional instrumentation is needed to monitor ICC. The applicant has adopted this position. Since incore thermocouples are required for BWRs as specified in Revision 2 (December 1980) to Regulatory Guide 1.97, the Zimmer-1 applicant will be required to address incore thermocouples requirements as stipulated in the referenced Regulatory Guide and described in Item (4) of documentation required by NUREG-0737, Section II.F.2.

With regard to the types and locations of displays and alarms for reactor water level in the control room, the staff has reviewed them under TMI Task Item I.D.1 and found them to be acceptable. However, we are requiring an ongoing program of control room design review and improvement to be implemented in conjunction with the detailed control room design review specified in TMI Task Item I.D.1.

The staff has reviewed the submittal dated April 22, 1981 in response to TMI Item II.F.2 requirements and has concluded that Zimmer-1 plant is in partial compliance with TMI Item II.F.2. Based on the results of our review we will require the applicant to commit to:

(1) incorporation of incore thermocouples into the ICC monitoring system prior to June 1983 in accordance with Regulatory Guide 1.97; and

(2) providing documentation required by Section II.F.2 of NUREG-0737, addressing the inclusion of thermocouples in the final ICC monitoring system, on a schedule acceptable to the staff.

The above commitments are required prior to issuance of the operating license.

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

Item 5 Assurance of Proper Engineered Safety Features Functioning

Position

Review all safety-related valve positions, positioning requirements, and positive controls to assure that valves remain positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also, review related procedures, such as those for maintenance, testing, plant and system startup, and supervisory periodic (e.g., daily/shift checks) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.

Discussion and Conclusions

This item requires the applicant to assess the function of all valves in engineered safety feature systems and to evaluate the procedures associated with operation, maintenance, and surveillance of these valves to ensure their proper alignment for engineered safety features functioning.

In a letter dated April 22, 1981 from J. D Flynn (CG&E) to H. Denton (NRC), the applicant has stated that valve positioning requirements, positive controls, and test and maintenance procedures associated with engineered safety features systems have been reviewed for Zimmer. It was indicated that motor-operated valves in safety systems are normally maintained in a configuration such as to require the least number of valve automatic movements upon system actuation. The position of vital manual emergency core cooling system valves is controlled by the use and documentation of locks on valve handwheels. Surveillance and testing procedures for engineered safety features systems include checklists to ensure the system is returned to standby upon completion of testing. Documentation is required upon return to service. The applicant has committed to confirm prior to fuel load that all engineered safety features systems are aligned in accordance with approved mechanical and electrical checklists. We consider the applicant's response to this item acceptable since confirmation of valve alignments will be provided before fuel load, and that our Office of Inspection and Enforcement will verify. We will condition the operating license for Zimmer for the verification.

Item 10 Safety-Related System Operability Status Assurance

Position

Review and modify, as required, procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known.

Discussion and Conclusions

As part of CG&E's response to IE Bulletin 79-08, it verified the review of procedures for removing from and restoring to service safety-related systems. The applicant's response is being reviewed by our Office of Inspection and Enforcement, Region III. We will condition the operating license for Zimmer for the resolution.

Item 22 Proper Functioning of Heat Removal Systems

Position

Describe the actions, both automatic and manual, necessary for proper functioning of the auxiliary heat removal systems (e.g., reactor core isolation cooling) that are used when the main feedwater system is not operable. For any manual action necessary, describe in summary form the procedure by which this action is taken in a timely sense.

Discussion and Conclusions

This item requires the applicant to review all actions needed to initiate and operate systems used for heat removal when the main feedwater system is not operable. For boiling water reactors, this may involve the use of the reactor core isolation cooling system, the steam condensing mode of the residual heat removal system, the high pressure core spray system, the safety relief valves, and the suppression pool cooling mode of the residual heat removal system.

In a letter dated April 22, 1981 from J. D. Flynn (CG&E) to H. Denton (NRC), the applicant described the actions necessary for proper functioning of the auxiliary heat removal systems that are used when the main feedwater is unavailable. The applicant indicated that if the main feedwater system is not operable, a reactor scram will be automatically initiated when reactor water level falls to Level 3 (529 inches above vessel bottom or 175 inches above the top of the active fuel). The operator can then remote manually initiate the reactor core isolation cooling system from the main control room, or the system will be automatically initiated when reactor water level decreases to Level 2 (479 inches above vessel bottom or 124 inches above the top of the active fuel) due to boil-off. At this point, the high pressure core spray system will also automatically start supplying makeup water to the vessel. These systems will continue automatic injection until the reactor water level reaches Level 8 (571 inches above vessel bottom or 217 inches above top of the active fuel), at which time the high pressure core spray injection valve is automatically closed, and the reactor core isolation cooling turbine is automatically tripped.

In the nonaccident case, the reactor core isolation cooling system is used to furnish subsequent makeup water to the reactor pressure vessel. Reactor core isolation cooling must be manually restarted (once it is tripped by a Level 8 signal) from the main control room by reopening the stop valve at the turbine inlet. (Manual restarting of RCIC will be corrected by Item II.K.3.13.) If the operator fails to restart reactor core isolation cooling, the high pressure core spray system will restart automatically when the level again falls to Level 2. No manual actions are required for high pressure core spray to start or restart. Reactor vessel pressure is regulated by the automatic or remote manual operation of the main steam relief valves which blow down to the suppression pool.

To remove decay heat, assuming that the main condenser is not available, the steam condensing mode of the residual heat removal system is initiated by the operator. This involves remote manual alignment of the residual heat removal system valves. If the steam condensing mode is unavailable for any reason, the main steam relief valves can be manually actuated from the control room. Remote manual alignment of the residual heat removal system into the suppression pool cooling mode is then required for suppression pool heat removal. Makeup water is still supplied by the reactor core isolation cooling system under manual control.

For the accident case with the reactor pressure vessel at high pressure, the high pressure core spray system is utilized to automatically provide the required makeup flow. No manual operations are required since the high pressure core spray system will cycle on and off automatically as water level reaches Level 2 and Level 8, respectively. If the high pressure core spray system fails under these conditions, the operator can manually depressurize the reactor vessel using the automatic depressurization system to permit the low pressure emergency core cooling systems to provide makeup coolant. Automatic depressurization will occur if all of the following signals are present: high drywell pressure, Level 1 water level (387 inches above vessel bottom or 32 inches above the top of the active fuel), pressure in at least one low pressure injection system and the runout of a 2-minute timer which starts with the coincidence of the other three signals. The operator may reset the timer to prevent depressurization.

The actions described by the applicant for operation of the auxiliary heat removal systems include manual restart (after Level 8 trip) of the reactor core isolation cooling system. Item II.K.3, item 13, of this report requires that the reactor core isolation cooling initiation logic be modified so that the system will restart automatically. The applicant has agreed to add the modifications for RCIC restart. Therefore, the applicant meets our present requirements as specified by this item, and is acceptable. We will condition the operating license for Zimmer for the review of the modification.

Item 23 Reactor Vessel Level Instrumentation

Position

Describe all uses and types of reactor vessel level indication for both automatic and manual initiation of safety systems. Describe other instrumentation that might give the operator the same information on plant status.

Discussion and Conclusions

In the NUREG-0737 response, April 22, 1981, the applicant summarizes the reactor vessel level instrumentation used at Zimmer. The instruments that sense the water level are differential pressure devices calibrated for accuracy at a specific vessel pressure and liquid temperature condition. This instrumentation is extensively detailed in the General Electric Company Report NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," and has been reviewed by us and evaluated in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications." We find this acceptable for this item.

II.K.3 Final Recommendations of Bulletins and Orders Task Force

Item 3 Failure of PORV or Safety Valve to Close

Position

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report. This requirement is to be met before fuel load.

Discussion and Conclusion

Since Zimmer has not yet operated, no valve failures have yet been reported. The applicant has committed to prompt reporting of safety/relief valve failures via the Licensee Event Report system and will summarize failures in the annual report. The plant Technical Specifications will require these failures to be reported within 30 days. We find this acceptable.

Item 13 Separation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation Levels

Position

Currently, the reactor core isolation cooling (RCIC) system and the high pressure coolant injection (HPCI) system both initiate on the same low water level signal and both isolate on the same high water level signal. The HPCI system will restart on low water level but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC system should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the RCIC system initiation logic should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analysis.

Discussion and Conclusions

At Zimmer, the high pressure core spray (HPCS) and RCIC are both initiated at low-water level Level 2 (479 inches above vessel zero or 124 inches above the top of the active fuel). Zimmer does not employ an HPCI system.

As a generic item, the possible separation of initiation levels for RCIC and HPCS was studied by General Electric for the BWR Owners Group. The applicant has endorsed the conclusions of that study and taken the position that the proposed separation of RCIC and HPCS initiation is unnecessary for safety considerations. The applicant based his conclusions on the following: for rapid level changes associated with accident scenarios and severe transients, their initiation would be essentially simultaneous in that possible separation distances could not preclude HPCS challenges; likewise, for slow level changes due to small leaks or slow transients, adequate time exists for manual initiation of RCIC by the reactor operator, prior to HPCS auto-initiation.

With regard to reducing thermal stresses on the vessel from cold water injection, the applicant discussed results of thermal fatigue analyses for BWR/3 and BWR/4 designs and indicated that these studies were bounding for BWR/5 and BWR/6. Thermal fatigue analyses show that the limiting reactor component is the feedwater nozzle for all plants equipped with HPCI and RCIC systems. The feedwater sparger is exposed to thermal cycles resulting from HPCI and RCIC operation as well as feedwater temperature changes during daily and weekly power swings. HPCS and RCIC injection locations on plants that do not inject through the feedwater system, as is the case for Zimmer, are not exposed to temperature variations during daily and weekly power swings, and hence see fewer thermal cycles. Likewise, changing the HPCS or RCIC initiation levels would not significantly impact the most limiting component, the feedwater nozzle, because of the separate injection points.

With regard to automatic restart of the RCIC system on low water level, the applicant has provided a satisfactory conceptual design for this capability. Remote manual reopening from the control room of the RCIC turbine steam supply trip and throttle valve is currently required to permit restart of the system.

We conclude that for Zimmer, the separation of HPCS and RCIC inititation levels is unnecessary at this time but note that the applicant is subject to the results of the ongoing generic evaluation of this topic. We find the conceptual design for the automatic restart of RCIC on low water level to be acceptable. The applicant has proposed a delay of installation to the first refueling outage based upon a forecast of equipment availability. We require further justification for this delay (equipment ordered, lead times, etc.) or a commitment for installation consistent with the schedule in NUREG-0737 (four months prior to the operating license). With the above justification or commitment, we conclude that the applicant will meet the requirements of this item.

Item 15 <u>Modify Break Detection Logic to Prevent Spurious Isolation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System</u>

<u>Position</u>

The high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. The pipe break detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems due to the pressure spike which accompanies startup of the systems. The pipe break detection circuitry should be modified so that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation

Submit sufficient documentation to support a reasonable assurance finding by the NRC that the modifications, as implemented, have resulted in satisfying the concerns expressed in the previous requirements.

Discussion and Conclusions

In a letter dated April 30, 1981 from J. D. Flynn (CG&E) to H. Denton (NRC), the applicant identified a circuit modification to assure that transients seen by pressure instruments used to sense flow in these two systems actually sense continuous high steam flow. Redundant Class 1E adjustable time delay relays are to be added to the logic of the RCIC and, because there is no turbine in the high pressure core spray (HPCS) system, there are no excess flow isolations or steam supply isolations as occurs in plants equipped with HPCI. We find the applicant's response acceptable.

We require the applicant to complete the modifications four months prior to issuance of operating license.

Item 16 Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification

Position

Failure of the power-operated relief valve to reclose during the TMI-2 accident resulted in damage to the reactor core. As a consequence, relief valves in all plants, including boiling water reactors, are being examined with a view toward their possible role in a small break loss-of-coolant accident.

The safety/relief valves are dual-function pilot-operated relief valves that use a spring-actuated pilot for the safety function and an external air-diaphragm-actuated pilot for the relief function.

The operating history of the safety/relief valves has been poor. A new design is used in some plants, but the operational history is too brief to evaluate the effectiveness of the new design. Another way of improving the performance of the valves is to reduce the number of challenges to the valves. This may be done by the methods described above or by other means. The feasibility and contraindications of reducing the number of challenges to the valves by the various methods should be studied. Those changes which are shown to decrease the number of challenges without compromising the performance of the valves or other systems should be implemented.

Results of the evaluation shall be submitted by April 1, 1981 for staff review. Documentation of the staff approved modification will be provided by January 1, 1982.

The actual modification will be accomplished during the next scheduled refueling outage after January 1, 1982 (if required).

Discussion and Conclusions

The applicant is a participant in the ongoing evaluation by the BWR Owners Group of possible ways to reduce challenges to safety/relief valves. That study encompasses the direct-acting Crosby safety/relief valve which is used at Zimmer.

The applicant has provided the results of the evaluation as prescribed by the position above. Based on the applicant's discussion, we conclude that no modifications are necessary for Zimmer at this time. We note, however, that the applicant is subject to the results of our generic review of this item.

Item 17 Report on Outages of Emergency Core Cooling Systems Licensee Report and Proposed Technical Specification Changes

Position

Several components of the emergency core cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the high pressure core injection system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Clarification

The present technical specifications contain limits on allowable outage times for ECC systems and components. However, there are no cumulative outage time limitations on these same systems. It is possible that ECC equipment could meet present technical specification requirements but have a high unavailability because of frequent outages within the allowable technical specifications.

The licensees should submit a report detailing outage dates and length of outages for all ECC systems for the last 5 years of operation, including causes of the outages. This report will provide the staff with a quantification of historical unreliability due to test and maintenance outages, which will be used to determine if a need exists for cumulative outage requirements in the technical specifications.

Based on the above guidance and clarification, a detailed report should be submitted. The report should contain (1) outage dates and duration of outages; (2) cause of the outage; (3) ECC systems or components involved in the outage; and (4) corrective action taken. Test and maintenance outages should be included in the above listings which are to cover the last 5 years of operation. The licensee should propose changes to improve the availability of ECC equipment, if needed.

Applicant for an operating license shall establish a plan to meet these requirements.

Discussion and Conclusions

Since Zimmer has not yet operated, there is no history of emergency core cooling system outages. However, by letter from James D. Flynn (CG&E) to H. Denton (NRC) dated May 1, 1981, the applicant has commited to report emergency core cooling system outages via the Licensee Event Report System and a summary of outages in the Annual Report. This satisfies our requirement for this item and is acceptable.

Item 18 Modification of Automatic Depressurization System Logic--Feasibility for Increased Diversity for Some Event Sequences

Position

The automatic depressurization system (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor vessel water level provided no high pressure coolant injection or high pressure core spray flow exists and a low pressure emergency core cooling (ECC) system is running. This logic would complement, not replace, the existing ADS actuation logic.

Discussion and Conclusion

The applicant is a participant in the BWR Owners Group study on this item. The applicant has committed to install the modifications during the first refueling outage. However, we require the applicant to submit the proposed modifications four months prior to issuance of operating license for staff review. We will review the proposed modifications when available, and will provide our conclusions in a supplement to this report. In the interim, should rapid vessel depressurization be required due to a break outside containment or a stuck open relief valve, manual actuation of the automatic depressurization system can be accomplished.

Item 21 Restart of Core Spray and Low Pressure Core Injection (LPCI) System

Position

The core spray and LPCI system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart if required to assure adequate core cooling. Because this design modification affects several core cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

Part a

By January 1, 1981, each licensee shall submit proposed design modifications and supporting analysis which will contain sufficient information to support a reasonable assurance finding by the NRC that the above position is met. The documentation should include as a minimum:

1. A discussion of the design with respect to the above paragraphs of Institute of Electronics and Electrical Engineers Standard 279-1971;

2. Support information including system design description, logic diagrams, electrical schematics, piping and instrument diagrams, test procedures, and technical specifications; and

3. Sufficient documentation to demonstrate that the systems, as modified, would not degrade proper system functions.

art b

icensee to implement modifications at the next refueling outage following staff approval of the design unless this outage is scheduled within 6 months of the approval date. In this event, modifications will be completed during the following refueling outage.

Discussion and Conclusions

In a letter dated April 22, 1981 from J. D. Flynn (CG&E) to H. Denton (NRC), the applicant discussed the BWR Owners Group evaluation of the feasibility of automatically restarting the low pressure core spray and low pressure coolant injection system on low water level. The applicant concluded that modifications for auto-restart of these systems are unnecessary for the following reasons: (1) the changes would increase system complexity and decrease system reliability, (2) current operator training emphasizes water level control, (3) emergency procedure guidelines emphasize water level control, (4) multiple control room indications of vessel water level will alert the operator to low level situations, and (5) the amount of time the operator has to correct errors is substantial.

With regard to item 5, the applicant presented the results of the BWR Owners Group analyses which indicate that for large recirculation line breaks, a minimum of 15 minutes is available after emergency core cooling system (ECCS) shutoff (assumed when water level reaches the top of the jet pumps) until peak fuel cladding temperatures reach 2200 degrees Fahrenheit. Time would be longer if the ECCS flow were terminated at a higher level or if other makeup systems such as the control rod drive or feedwater flows were operable.

Based on the applicant's discussion of the BWR Owners Group analyses and recognizing the emphasis placed on water level control in boiling water reactor operator training. we agree that no modification to provide for automatic restart of the low pressure ECCS system is necessary for Zimmer, and find the response addressing low pressure systems to be acceptable.

With regard to high pressure core spray (HPCS) restart, the applicant has committed [Letter dated May 1, 1981 from J. D. Flynn (C.G.&E) to H. Denton (NRC)] to provide an auto start for the high pressure core spray, consistent with the modification proposed by the BWR Owners Group (letter dated December 29, 1980 from Waters (General Electric) to Eisenhut (NRC)). The applicant is also committed to meeting the schedule defined in Part (b) of NUREG-0737, Item II.K.3.21. This commitment satisfies the requirements of this item. The plant emergency procedures will provide reasonable assurance of safety until the installation of the automatic equipment.

Item 22 Automatic Switchover of Reactor Core Isolation Cooling System Suction--Verify Procedures and Modify Design

Position

The reactor core isolation cooling (RCIC) system takes suction from the condensate storage tank with manual switchover to the suppression pool when the condensate storage tank level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC system suction from the condensate storage tank to the suppression pool.

Discussion and Conclusion

Zimmer has an automatic switch-over from the condensate storage tank to the suppression pool for the RCIC suction and therefore meets the requirements of this position. We conclude that no modifications are necessary.

Item 24 Confirm Adequacy of Space Cooling for Reactor Core Isolation Cooling and High Pressure Coolant Injection Systems

Position

Long-term operation of the reactor core isolation cooling and high pressure coolant injection systems may require space cooling to maintain the pump-room temperatures within allowable limits. Applicants should verify the acceptability of the consequences of a complete loss of alternating-current power. The reactor core isolation cooling and high pressure coe injection systems should be designed to withstand a complete loss of offsite alternating-current power to their support systems, including coolers, for at least 2 hours.

Discussion and Conclusions

The applicant's response to this requirement states that the Zimmer plant utilizes an integral heat-recovery HVAC concept for normal operations. In addition to this, the plant utilizes a cubicle arrangement for separation purposes. Each ECCS equipment room has an independent emergency area cooling unit. These cooling units are not redundant. They are sized for abnormal and accident conditions to maintain ECCS equipment within allowable limits (148 deg. F) following a LOCA.

Power is supplied from essential power buses with control circuits energized from the same essential bus. Instrument power is from Class IE sources. Divisionalization of ECCS functions includes the essential power to the corresponding ECCS functions includes the essential power to the corresponding ECCS emergency area cooling unit. This makes each subsystem independent and because each ECC system has a redundant functional equivalent, the loss of a particular ECCS or its cubicle or its equipment area cooling system, does not preclude the essential safety function. In such a case, the essential safety function is accomplished by auto-initiation of the redundant ECCS in the counterpart cubicle.

Evaluation of adequacy of ECCS systems such as HPCS at Zimmer, therefore, is already represented in the ECCS analysis of FSAR Chapter 6.0 because the space coolers are part of the ECC system itself. Design adequacy is confirmed via performance evaluations associated with preoperational and start-up testing of the individual ECC systems with normal power sources.

FSAR Subsection 15.1.19 addresses the total loss of AC power to the station and no threat to the public health and safety ensues.

The staff finds the response adequate except that final confirmation will be made during the staff's review of the applicant's Environmental Qualification Program (see subsection 3.11 of this supplement).

Item 25 Effect of Loss of Alternating Current Power on Pump Seals

Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current power for at least 2 hours. Adequacy of the seal design should be demonstrated. The results of the valuation and proposed modifications are due by July 1, 1981. Modificat are to be implemented by January 1, 1982.

Clarification

The intent of this position is to prevent excessive loss of reactor coolant system inventory following an anticipated operational occurrence. Loss of alternating current power for this case is construed to be loss of offsite power. If seal failure is the consequence of loss of cooling water to the reactor coolant pump seal coolers for 2 hours, due to loss of offsite power, one acceptable solution would be to supply emergency power to the component cooling water pump.

Discussion and Conclusions

Support systems necessary to cool the recirculation pump seals (i.e., service water system and reactor building closed cooling water system) are supplied from an emergency diesel generator bus. If either cooling system fails to operate, the other system will provide adequate cooling to the pump seals to prevent damage. This satisfies our current requirements for this item.

Item 27 Provide Common Reference Level for Vessel Level Instrumentation

Position

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point. Either the bottom of the vessel or the top of the active fuel are reasonable reference points.

Discussion and Conclusions

The applicant's response to this requirement states that a decription of the various water level ranges, calibration data, and the conditions under which each range is used can be found in FSAR Subsection 7.6.1.2.3.1.2 and the regions of the vessel they cover can be found in NEDO-24708A "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors" Section 2.3.2 "Reactor Water Level Instrumentation".

The narrow range, wide range, shutdown range, and upset range water level instruments are all referenced to a point near the bottom of the steam dryer skirt. The fuel zone range instrument is referenced to a point near the top of the active fuel. This arrangement provides a common water level reference point for all normal operating and accident conditions except a large-break

LOCA. In the case as the primary source of water level indication and verification of core reflood. This difference in reference point is not confusing to the operator due to training and experience. The operator's awareness of this difference is constantly reinforced during routine control room surveillance since the fuel zone level instrument is always off-scale high while adjacent water level instrumentation is reading on scale.

The current arrangement and reference points for the water level instrumentation will allow the operators to make timely and correct decisions regarding reactor inventory makeup requirements. Identification of a common water level reference is not vital to ensure safe reactor operation.

Based on the above, it has been concluded tht the current ZPS water level instrumentation system is fully adequate to allow plant operators to respond properly under all postulated reactor conditions and that no modification of the current control room water level instrumentation is required on the basis of plant safety considerations.

The staff concludes that the applicant response does not meet the stated requirement since it does not provide for a common reference level for vessel level instrumentation. We will require the common reference level prior to fuel loading.

Item 28 Verify Qualification of Accumulators on Automatic Depressurization System Valves

Position

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. General Electric has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensee and applicant should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee and applicant must show that the accumulator design is still acceptable.

Clarification

The ADS valves, accumulators, and associated equipment and instrumentation must be capable of performing their functions during and following exposure to hostile environments and taking no credit for nonsafety-related equipment or instrumentation. Additionally, air (or nitrogen) leakage through valves must be accounted for in order to assure that enough inventory of compressed air is available to cycle the ADS valves.

Discussion and Conclusions

The applicant has committed to participate in the BWR Owner's Group which is conducting an evaluation of the adequacy of the ADS configurations with respect to the above requirements. The results of the analysis as it applies to the Zimmer plant are to be provided by January 1, 1982. This commitment meets our requirements for this item and, therefore, is acceptable.

Item 30 Revised Small Break Loss-of-Coolant Accident Methods to Show Compliance With 10 CFR Part 50, Appendix K

Position

The analysis methods used by nuclear steam supply system vendors and/or fuel suppliers for small break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Clarification

As a result of the accident at TMI-2, the Bulletins and Orders Task Force was formed within the Office of Nuclear Reactor Regulation. This task force was charged, in part, to review the analytical predictions of feedwater transients and small break LOCAs for the purpose of assuring the continued safe operation of all operating reactors, including a determination of acceptability of emergency guidelines for operators.

As a result of the task force reviews, a number of concerns were identified regarding the adequacy of certain features of small break LOCA models, particularly the need to confirm specific model features (e.g., condensation heat transfer rates) against applicable experimental data. These concerns, as they applied to each light water reactor (LWR) vendor's models, were documented in the task force reports for each LWR vendor. In addition to the modeling concerns identified, the task force also concluded that, in light of the TMI-2 accident, additional systems verification of the small break LOCA model as required by II.4 of Appendix K to 10 CFR Part 50 was needed. This included providing predictions of Semiscale Test S-07-10B, LOFT Test (L3-1), and providing experimental verification of the various modes of single-phase and two-phase natural circulation predicted to occur in each vendor's reactor during small break LOCAs.

Based on the cumulative staff requirements for additional small break LOCA model verification, including both integral system and separate effects verification, the staff considered model revision as the appropriate method for reflecting any potential upgrading of the analysis methods.

The purpose of the verification was to provide the necessary assurance that the small break LOCA models were acceptable to calculate the behavior and consequences of small primary system breaks. The staff believes that this assurance can alternatively be provided, as appropriate, by additional justification of the acceptability of present small break LOCA models with regard to specific staff concerns and recent test data. Such justification could supplement or supersede the need for model revision.

The specific staff concerns regarding small break LOCA models are provided in the analysis sections of the B&O Task Force reports for each LWR vendor. These concerns should be reviewed in total by each holder of an approved emergency core cooling system model and addressed in the evaluation as appropriate.

The recent tests include the entire Semiscale small break test series and LOFT Tests (L3-1) and (L3-2). The staff believes that the present small break LOCA models can be both qualitatively and quantitatively assessed against these tests. Other separate effects tests (e.g., Oak Ridge National Laboratory core uncovery tests) and future tests, as appropriate, should also be factored into this assessment.

Based on the preceding information, a detailed outline of the proposed program to address this issue should be submitted. In particular, this submittal should identify (1) which areas of the models, if any, the licensee intends to upgrade, (2) which areas the licensee intends to address by further justification of acceptability, (3) test data to be used as part of the overall verification/upgrade effort, and (4) the estimated schedule for performing the necessary work and submitting this information for staff review and approval.

Discussions and Conclusions

The applicant is a participant with the BWR Owners Group which is reviewing the Appendix K methodology. The applicant has indicated that any model improvements will be utilized for plant-specific reanalyses, after approval of models by us, if required. In the interim, the current emergency core cooling system evaluation model, which has been approved by us, is considered sufficiently conservative for licensing of Zimmer.

Item 31 Plant-Specific Calculations to Show Compliance with 10 CFR Part 50.46

Position

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents as described in II.K.3 item 30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

Calculations to be submitted by January 1, 1983 or 1 year after staff approval of loss-of-coolant accident analysis models, whichever is later (required only if model changes have been made).

Discussion and Conclusions

The applicant has committed to providing plant-specific calculations if the results of II.K.3 item 30 indicate this need. This commitment meets current requirements and is acceptable. In the interim, the current plant-specific analysis is considered to be sufficiently conservative to permit licensing of Zimmer.

Item 44 Evaluation of Anticipated Transients With Single Failure To Verify No Fuel Failure

Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result in a stuck-open relief valve should be included in this category. The results of the evaluation are due January 1, 1981.

Discussion and Conclusions

In a letter dated May 1, 1981 from J. D. Flynn (CG&E) to H. Denton (NRC), the applicant has provided information discussing an evaluation performed by the BWR Owners Group. The applicant has stated that the study results show that adequate core cooling is maintained for any transient with the worst single failure. The bounding event for Zimmer was stated to be the loss of feedwater transient with concurrent failure of the high pressure emergency core cooling system. In this case, the core always remained covered. The applicant also referenced studies involving a stuck-open safety/relief valve in addition to the worst transient and worst single failure. The results indicated that the core remained covered and adequate core cooling was available during the course of the transient.

The applicant has committed to verify that Zimmer is bounded by the GE generic analysis. The applicant has also committed to provide a summary of operator actions required to accomplish hot shutdown during the worst case event. We will report our findings in a supplement to this report.

Item 45 Evaluation of Depressurization With Other Than Automatic Depressurization System

Position

Analyses to support depressurization modes other than full actuation of the automatic depressurization system (e.g., early blowdown with one or two safety/relief valves) should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown.

Discussion and Conclusion

The applicant reported the results of the BWR Owners Group study in a letter dated April 22, 1981 from J. D. Flynn (CG&E) to H. Denton (NRC). The analyses assumed failure of all high pressure injection systems but operability of all low pressure systems. The time at which the operator is assumed to actuate the automatic depressurization system varied. The effects of depressurization over a 10-minute interval and a 20-minute interval were compared to the full blowdown case which is completed in 3.3 minutes. The key parameter studied in regard to vessel integrity was vessel fatigue usage. The potential for a reduction in fatigue usage as a result of a longer blowdown period was examined relative to the impact on core cooling capability. The applicant concluded that:

(1) Vessel integrity limits are not exceeded for full automatic depressurization system blowdown.

(2) For slower depressurization rates (longer than the approximate 3.3 minute interval associated with the normal depressurization rate), there is little impact on vessel fatigue usage relative to that usage assignable to the full automatic depressurization system blowdown.

(3) Slower depressurization rates have an adverse impact on core cooling capability except when the operator initiates blowdown very early in the accident.

The results indicate that some improvement in core cooling capability was possible using a 10-minute blowdown period if the operator actuated the automatic depressurization system within 1-6 minutes after the initiation of the accident. The applicant indicated that during this initial time period it would be more prudent to attempt to activate the high pressure injection systems in an effort to avoid use of the automatic depressurization system.

Based on the applicant's discussion, we conclude that no change to the current mode of depressurization is necessary for Zimmer at this time. We note, however, that the applicant is subject to the results of our generic review of this item.

Item 46 Response to List Michelson's Concerns

Position

General Electric should provide a response to the Michelson concerns as they relate to boiling water reactors.

Clarification

General Electric provided a response to the Michelson concerns as they relate to boiling water reactors by letter dated February 21, 1980. Licensees and applicants should assess applicability and adequacy of this response to their plants.

Discussion and Conclusions

The applicant is a participant of a BWR Owners Group which has completed an evaluation of this matter. In a letter dated May 1, 1981 from James D. Flynn (CG&E) to H. Denton (NRC), the applicant has indicated that the General Electric Company's generic response to the Michelson concerns is applicable to Zimmer. The generic responses were forwarded by letters dated Fabruary 21, 1981 (Buchholz from General Electric to Eisenhut (NRC)). We reviewed the generic responses for their applicability to Zimmer and agree with the applicant that they do apply. No changes or modifications are necessary for Zimmer at this time. We note, however, that the applicant is subject to the results of our generic review of this item.

III. Emergency Preparations and Radiation Protection

III.A.1.1 Upgrade Emergency Preparedness

Position

Provide an emergency response plan in compliance with NUREG-0654, Rev. 1 (November 1980) "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." NRC will give substantial weight to FEMA findings on offsite plans in judging the adequacy against NUREG-0654. Perform an emergency exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations.

This requirement shall be met before issuance of a full-power license.

Discussion and Conclusions

We have reviewed the applicant's revised emergency plan against the current regulatory requirements contained in 10 CFR Part 50 and the guidance criteria in NUREG-0654 dated November 1980. Upon satisfactory completion of the items identified below, the staff will issue a finding with respect to emergency preparedness matters for full power operation of the Zimmer Nuclear Generating Station.

1. Correct the deficiencies identified in our Emergency Preparedness Evaluation Report which is included as Appendix G to this report (NUREG-0694, item III.A.1.1).

2. Perform an emergency response exercise that tests the integrated capability and a major portion of the basic elements existing within the emergency preparedness plans and organizations (NUREG-0694, item III.A.1.1).

3. Submit revised radiological response plans of State and local governments within the plume exposure pathway Emergency Planning Zone as well as the plans of State governments within the ingestion pathway Emergency Planning Zone that conform to the criteria of NUREG-0654 (10 CFR 50.33g).

III.A.1.2 Upgrade Emergency Support Facilities

Position

Provide radiation monitoring and ventilation systems, including particulate and charcoal filters, and otherwise increase the radiation protection to the onsite technical support center to assure that personnel in the center will not receive doses in excess of 5 rem to the whole body or 30 rem to the thyroid for the duration of the accident. Provide direct display of plant safety system parameters and call up display of radiological parameters.

For the near-site Emergency Operations Facility, provide shielding against direct radiation, ventilation isolation capability, dedicated communications with the onsite Technical Support Center and direct display of radiological and meteorological parameters.

Discussion and Conclusions

The above requirements are those set forth in NUREG-0696, "Functional Criteria for Emergency Response Facilities," dated February 1981 which specifies the functional criteria necessary for the design and implementation of the Technical Support Center and Emergency Operations Facility.

The Emergency Plan, dated January 1981, describes the Technical Support Center and the near-site Emergency Operations Facility that have been established on an interim basis. As a result of our review and the applicant's commitments to meet the requirements set forth in NUREG-0696, we conclude that these interim facilities are acceptable for full power operation pending their upgrading to meet the NUREG-0696 requirements.

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

Position

1. Each nuclear facility shall upgrade its emergency plan to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654.

2. Perform an emergency response exercise to test the integrated capability and a major portion of the basic elements existing within emergency preparedness plans and organizations.

Discussion and Conclusions

By letter dated January 1, 1981, the applicant submitted upgraded emergency plans for the Zimmer site and the State and local entities within the plume and ingestion exposure emergency planning zones. We reviewed the applicant's onsite emergency plan and prepared an Emergency Preparedness Evaluation Report which is attached to this report (see Appendix G of this report). The Evaluation Report lists each planning standard of 10 CFR Part 50.47(b), followed by a discussion of how the applicant meets the standard. Deficiencies with respect to the regulation are identified.

After receiving the findings and determinations made by FEMA on the State and local emergency response plans, and after reviewing the revised applicant's plan, a supplement to this Safety Evaluation Report will provide our conclusions on the status of emergency preparedness for Zimmer and related emergency planning zones.

Based on our review, we conclude that the Zimmer Site Emergency Plan, upon satisfactory corrections of the open items listed in Appendix G of this report, will meet the planning standards of 10 CFR 50.47(b) and conform to the guidance stated in NUREG 0654, Revision 1. We will address the resolution of these items in a supplement to this report.

III.D Radiation Protection

III.D.1.1 Integrity of Systems Outside Containment Likely to Contain Radioactive Material for Pressurized-water Reactors and Boiling-water Reactors

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

1. Immediate leak reduction

   a. Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

b. Measure actual leakage rates with system in operation and report them to the NRC.

2. Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic inategrated leak tests at intervals not to exceed each refueling cycle.

Clarification

Applicants shall pro ide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident.

1. Systems that should be leak tested are as follows (any other plant system which has similar functions or postaccident characteristics even though not specified herein, should be included):

   a. Residual heat removal,

   b. Containment spray recirculation,

   c. High-pressure injection recirculation,

   d. Containment and primary coolant sampling,

   e. Reactor core isolation cooling,

   f. Waste gas (includes headers and cover gas system outside of containment in addition to decay or storage system).

   Include a list of systems containing radioactive materials which are excluded from program and provide justification for exclusion.

2. Testing of gaseous systems should include helium leak detection or equivalent testing methods.

3. Should consider program to reduce leakage potential release paths due to design and operator deficiencies as discussed in our letter to all operating nuclear power plants regarding North Anna and Related incidents, dated October 17, 1979.

4. This requirement shall be implemented by applicants for operating license prior to issuance of a full-power license.

Discussion and Conclusions

In a letter dated April 22, 1981, Cincinnati Gas and Electric committed to a program to reduce leakage from systems outside of containment prior to the issuance of a full-power license, in order to satisfy the requirement of III.D.1.1. The systems to be initially tested, prior to full-power operation and the applicable method of obtaining actual leak rates have been specified. The commitment addresses the North Anna and related incidents letter and provides a continuing leak reduction program frequency.

We find the proposed program to reduce leakage from systems outside of containment meets the requirements for III.D.1.1 given above and in NUREG-0578, 0660, and 0694; therefore, the program is acceptable.

III.D.3.3 Improved Inplant Iodine Instrumentation Under Accident Conditions

Position

1. Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

2. Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Clarification

Effective monitoring of increasing iodine levels in the buildings under accident conditions must include the use of portable instruments using sample media that will collect iodine selectively over xenon (e.g., silver zeolite) for the following reasons:

1. The physical size of the auxiliary and/or fuel handling building precludes locating stationary monitoring instrumentation at all areas where airborne iodine concentration data might be required.

2. Unanticipated isolated "hot spots" may occur in locations where no stationary monitoring instrumentation is located.

3. Unexpectedly high background radiation levels near stationary monitoring instrumentation after an accident may interfere with filter radiation readings.

4. The time required to retrieve samples after an accident may result in high personnel exposures if these filters are located in high-dose-rate areas.

After January 1, 1981, each applicant and licensee shall have the capability to remove the sampling cartridge to a low-background, low-contamination area for further analysis. Normally, counting rooms in auxiliary buildings will not have sufficiently low backgrounds for such analyses following an accident. In the low background area, the sample should first be purged of any entrapped noble gases using nitrogen gas or clean air free of noble gases. The licensee shall have the capability to measure accurately the iodine concentrations present on these samples under accident conditions. There should be sufficient samplers to sample all vital areas.

For applicants with fuel loading dates prior to January 1, 1981, provide by fuel loading (until January 1, 1981) the capability to accurately detect the presence of iodine in the region of interest following an accident. This can be accomplished by using a portable or cart-mounted iodine sampler with attached single-channel analyzer (SCA). The SCA window should be calibrated to the

365 KeV of iodine-131 using the SCA. This will give an initial conservative estimate of presence of iodine and can be used to determine if respiratory protection is required. Care must be taken to assure that the counting system is not saturated as a result of too much activity collected on the sampling cartridge.

Discussion and Conclusion

Zimmer will have the capability of detecting and analyzing post-accident concentrations of radioiodine in the control room and other areas of the facility where personnel may be present during an accident. The Fixed Airborne Activity Monitoring System (FAAM) monitors the six major HVAC ducts leading to the plant vent stack. Three air sample panels permit plant personnel to obtain remote grab air particulate/iodine (P/I) samples from 25 rooms throughout the plant. Eleven portable air samplers are also available for obtaining grab air P/I samples throughout the plant. Silver Zeolite Iodine sample cartridges will be available for use with these systems at Zimmer in the presence of noble gases during accident conditions.

The applicant will have three portable Eberline Instrument Corporation Model SAM-2 SCA's available for analyzing grab samples. For a more thorough analysis, the applicant has a fixed counting facility containing single and multichannel analyzers. Direct radiation levels in this counting area under accident levels will be 0.015 Rad/hr during the 30-day postaccident period. The detectors are sufficiently shielded to permit the detection of concentrations of ioding below occupational MPC. In addition, the detector caves can be purged with bottled nitrogen to preclude the admission of airborne contaminents. Bottled nitrogen is also available to purge iodine cartridges of noble gases. As suggested in NUREG-0737, the applicant will provide procedures and training necessary to accurately detect and analyze the presence of airborne radioiodine following an accident.

Status

The applicant has adequately addressed the criteria of Item III.D.3.3 and his response meets the positions set forth in NUREG-0737.

III.D.3.4 Control Room Habitability

Position

In accordance with Item III.D.3.4, "Control Room Habitability," applicants shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19 of General Design Criteria).

Clarification

1. All applicants must make a submittal to us regardless of whether or not they met the criteria of the referenced Standard Review Plan sections. The new clarification specifies that applicants that meet the criteria of the Standard Review Plans should provide the basis for their conclusion

that Section 6.4 of the Standard Review Plan requirements are met. Applicants may establish this basis by referencing past submittals to us and/or providing new or additional information to supplement past submittals.

2. All applicants with control rooms that meet the criteria of the following Sections of the Standard Review Plan:

2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity,
2.2.3	Evaluation of Potential Accidents, and
6.4	Habitability Systems

shall report their findings regarding the specific Standard Review Plan sections as explained below. The following documents should be used for guidance:

a. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of Regulatory Power Plant Control Room During a Postulated Hazardous Chemical Release;"

b. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accident Chlorine Release;" and

c. K. G. Murphy and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Design Criterion 19," 13th AEC Air Cleaning Conference, August 1974.

Applicants shall submit the results of their findings as well as the bases for those findings by January 1, 1981. In providing the basis for the habitability finding, applicants may reference their past submittals. Applicants should, however, ensure that these submittals reflect the current facility design and that the information requested in Table III.D.3.4-1 is provided.

TABLE III.D.3.4-1

INFORMATION REQUIRED FOR CONTROL ROOM HABITABILITY EVALUATION

(1) Control-room mode of operation, i.e., pressurization and filter recirculation for radiological accident isolation or chlorine release

(2) Control-room characteristics:

(a) air volume control room

(b) control-room emergency zone (control room, critical files, kitchen, washroom, computer room, etc.)

(c) control-room ventilation system schematic with normal and emergency air-flow rates

(d) infiltration leakage rate

(e) high efficiency particulate air filter and charcoal adsorber efficiencies

TABLE III.D.3.4-1 (Continued)

(f) closest distance between containment and air intake

(g) layout of control room, air intakes, containment building, and chlorine, or other chemical storage facility with dimensions

(h) control-room shielding including radiation streaming from penetrations, doors, ducts, stairways, etc.

(i) automatic isolation capability-damper closing time, damper leakage and area

(j) chlorine detectors or toxic gas (local or remote)

(k) self-contained breathing apparatus availability (number)

(l) bottled air supply (hours supply)

(m) emergency food and potable water supply (how many days and how many people)

(n) control-room personnel capacity (normal and emergency)

(o) potassium iodide drug supply

(3) Onsite storage of chlorine and other hazardous chemicals:

(a) total amount and size of container

(b) closest distance from control-room air intake

(4) Offsite manufacturing, storage, or transportation facilities of hazardous chemicals

(a) identify facilities within a 5-mile radius;

(b) distance from control room

(c) quantity of hazardous chemicals in one container

(d) frequency of hazardous chemical transportation traffic (truck, rail, and barge)

(5) Technical specifications (refer to standard technical specifications)

(a) chlorine detection system

(b) control-room emergency filtration system including the capability to maintain the control-room pressurization at 1/8-inch water gauge, verification of isolation by test signals and damper closure times, and filter testing requirements.

3. All applicants with control rooms that do not meet the criteria of the above-listed references, Standard Review Plans, regulatory guides, and other references shall perform the necessary evaluations and identify appropriate modifications.

Each applicant submittal shall include the results of the analyses of control room concentrations from postulated accidental release of toxic gases and control room operator radiation exposures from airborne radioactive material and direct radiation resulting from design basis accidents. The toxic gas accident analysis should be performed for all potential hazardous chemical releases occurring either on the site or within 5 miles of the plant site boundary. Regulatory Guide 1.78 lists the chemicals most commonly encountered in the evaluation of the control room habitability but is not all inclusive.

The design basis accident radiation source term should be for the loss-of-coolant accident containment leakage and engineered safety features leakage contribution outside containment as described in Appendices A and B in Section 15.6.5 of the Standard Review Plan. In addition, boiling-water reactor facility evaluations should add any leakage from the main steam isolation valves (i.e, valve steam leakage, valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and engineered safety features leakage following a loss-of-coolant accident. This should not be construed as altering our recommendations in Section D of Regulatory Guide 1.95 (Rev. 2) regarding main steam isolation valve leakage-control systems. Other design basis accidents should be reviewed to determine whether they might constitute a more severe control room hazard than the loss-of-coolant accident.

In addition to the accident analysis results, which should either identify the possible need for control room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control room operators to remain in the control room to take appropriate actions required by Criterion 19 of the General Design Criteria, the applicant should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Table III.D.3.4-1 lists the information that should be provided along with the applicant's evaluation.

Discussion and Conclusions

The staff advised the applicant of the control room habitability requirements that must be met prior to the issuance of a full power license for the Zimmer Unit 1 facility. These requirements are identified in NUREG-0660 (May 1980), "NRC Action Plan Developed as a Result of TMI-2 Accident," in NUREG-0694 (June 1980), "TMI Related Requirements for New Operating Licenses" and in NUREG-0737 (October 1980), "Clarification of TMI Action Plan Requirements."

The applicant responded to NUREG-0694 in a letter dated November 26, 1980 providing a status of the ongoing evaluation of control room habitability under Item III.D.3.4. In a letter dated April 22, 1981, the applicant submitted an advance copy of a response to NUREG-0737 to be included as Appendix L to the FSAR in May 1981. The applicant states that the control room habitability requirements for the Zimmer Unit 1 have been met.

The staff evaluation of the control room habitability aspect is presented in Section 6.4 of the Safety Evaluation Report (SER) of January 1979 and in

Section 6.4.2 of this Supplement to the SER. The staff finds in these sections that the control room operators will be adequately protected against the effects of an accidental release of radioactive gases and of the toxic gases chlorine and ammonia in the vicinity of the site. The staff concludes that the Zimmer Unit 1 control room habitability system meets the requirements of General Design Criterion 19 of 10 CFR Part 50 Appendix A and the guidelines of Regulatory Guides 1.78 and 1.95.

The staff evaluation of the location, type, and size of potential toxic gas hazards in the vicinity of the site is presented in Section 6.4.2 of the SER of January 1979 and in Section 2.2 of this SER Supplement. The staff finds that chlorine and ammonia are potential toxic gas hazards in the vicinity of the site. However, the staff did not complete its evaluation of potential toxic gas hazards resulting from truck traffic on nearby U.S. Route 52. Should the continued staff evaluation of this aspect identify any potential toxic gas hazard in addition to chlorine and ammonia, then the staff will require the applicant to provide the appropriate additional toxic gas detection and protective capability.

23 CONCLUSIONS

Based on our evaluation of the application as set forth in NUREG-0528 and in this supplement, we are able to affirm the conclusions presented in Section 22.0 of NUREG-0528.

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APPENDIX A

CHRONOLOGY (Continued from NUREG-0528)
(Major Safety Review Correspondence)

December 22, 1978	Letter to CG&EC concerning interim report on motor operated valves.
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APPENDIX A (Continued)

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APPENDIX A (Continued)

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November 17, 1980	Letter to CG&EC regarding pressure isolation valves.
November 18, 1980	Letter to CG&EC requesting July 27, 1980 earthquake data.
November 19, 1980	Letter from CG&EC regarding masonry walls.
November 19, 1980	Letter from CG&EC regarding financial information.
November 20, 1980	Letter to CG&EC regarding plant staffing plans.
November 20, 1980	Letter to CG&EC regarding low power test program.
November 20, 1980	Letter to CG&EC regarding leak chase channels.
November 26, 1980	Letter from CG&EC forwarding response to NUREG-0694.
November 26, 1980	Letter to CG&EC regarding fracture toughness.
November 26, 1980	Letter to CG&EC regarding environmental qualification of equipment.
December 1, 1980	Letter from CG&EC regarding low pressure turbine discs.
December 3, 1980	Letter to CG&EC regarding soil structure analysis.
December 5, 1980	Letter from CG&EC transmitting Amendment 111, Revision 68 to the FSAR.

December 8, 1980	Letter from CG&EC regarding pressure boundary fracture toughness.
December 9, 1980	Letter to CG&EC forwarding Revision 1 to NUREG-0654.
December 9, 1980	Letter from CG&EC regarding seismic piping analysis.
December 19, 1980	Letter to CG&EC requesting additional financial information.
December 19, 1980	Letter from CG&EC transmitting Amendment 112, Revision 69 to the FSAR.
December 22, 1980	Letter to CG&EC regarding BWR scram discharge system.
December 22, 1980	Letter to CG&EC regarding NUREG-0612.
December 30, 1980	Letter to CG&EC.
January 7, 1981	Letter from CG&EC regarding fuel load date.
January 7, 1981	Letter from CG&EC forwarding Environmental Qualification of Safety-Related Electrical Equipment Report.
January 14, 1981	Letter to CG&EC regarding toxic chemical traffic on US 52.
January 14, 1981	Letter to CG&EC regarding Regulatory Guide 1.56, Revision 1.
January 14, 1981	Letter to CG&EC regarding low power test program.
January 14, 1981	Letter to CG&EC regarding exemptions to Appendices G and H.
January 19, 1981	Letter to CG&EC regarding environmental qualification of equipment.
January 20, 1981	Letter from CG&EC forwarding seismic qualification equipment.
January 26, 1981	Letter from CG&EC forwarding financial information.
January 27, 1981	Letter to CG&EC concerning control room review.
January 28, 1981	Letter from CG&EC concerning pressure isolation valves.
January 29, 1981	Letter to CG&EC regarding NEDO-24154.
January 29, 1981	Letter to CG&EC regarding containment pressure boundary materials.
January 30, 1981	Letter from CG&EC forwarding emergency plans.

APPENDIX A (Continued)

January 30, 1981	Letter to CG&EC regarding emergency procedures.
January 30, 1981	Letter to CG&EC regarding seismic qualification of equipment.
February 5, 1981	Letter from CG&EC regarding control room review.
February 5, 1981	Letter from CG&EC forwarding emergency plans.
February 6, 1981	Letter from CG&EC transmitting Amendment 113, Revision 70 to the FSAR.
February 10, 1981	Letter from CG&EC forwarding emergency operating procedures.
February 13, 1981	Letter to CG&EC regarding leakage from RHR suction valves.
February 19, 1981	Letter to CG&EC regarding snubber inspection.
February 25, 1981	Letter to CG&EC regarding station blackout event.
February 26, 1981	Letter to CG&EC regarding NUREG-0313, Revision 1.
February 27, 1981	Letter to CG&EC forwarding radiological emergency plans.
March 3, 1981	Letter to CG&EC regarding low pressure turbine discs.
March 5, 1981	Letter from CG&EC forwarding low power testing commitment.
March 5, 1981	Letter to CG&EC regarding NUREG-0696.
March 6, 1981	Letter from CG&EC regarding leak chase channels.
March 10, 1981	Letter to CG&EC regarding environmental qualification of equipment.
March 13, 1981	Letter to CG&EC regarding review schedules.
March 16, 1981	Letter to CG&EC regarding hydrogen control.
March 17, 1981	Letter to CG&EC regarding seismic design.
March 25, 1981	Letter to CG&EC regarding Q-List updating.
March 25, 1981	Letter from CG&EC forwarding Revision 11 to the Industrial Security Plan.
March 26, 1981	Letter from CG&EC regarding hydrogen control.

APPENDIX B

ACRS REPORT ON THE ZIMMER NUCLEAR POWER STATION

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

March 13, 1979

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON WILLIAM H. ZIMMER NUCLEAR POWER STATION, UNIT 1

Dear Dr. Hendrie:

During its 227th meeting, March 8-10, 1979, the Advisory Committee on Reactor Safeguards completed its review of the application of the Cincinnati Gas and Electric Company (CG&E), the Columbus and Southern Ohio Electric Company, and the Dayton Power and Light Company (hereinafter referred to collectively as the Applicants) for authorization to operate the William H. Zimmer Nuclear Power Station, Unit 1. CG&E will be responsible for operating the plant. A tour of the facility was made by members of the Subcommittee on November 16, 1978 and the application was considered at Subcommittee meetings on November 17, 1978 and February 27, 1979. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicants, the General Electric Company, Sargent and Lundy Company, Kaiser Engineers Incorporated and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for this plant on September 17, 1971.

The Zimmer Nuclear Power Station is located in Ohio on the Ohio River approximately 24 miles southeast of Cincinnati and one-half mile north of Moscow, Ohio. The plant will utilize a 2436 MWt BWR/5 boiling water reactor which is similar to the BWR/4 used in the Edwin I. Hatch Nuclear Plant, Unit No. 2. A principal difference is the use of recirculation flow control valves to regulate power rather than pump speed control which has been used on plants of the BWR/4 type.

The Zimmer Nuclear Power Station has a Mark II pressure suppression containment and is designated as one of the lead plants for this type containment. The NRC Staff has reviewed the generic aspects of the Mark II containment system and has reported its findings in NUREG-0487. The generic aspects of Mark II load evaluation and acceptance criteria were considered at Subcommittee meetings on July 7-8, 1977, November 30, 1977, May 23, 1978, and November 28-30, 1978. The Committee believes that the acceptance criteria are suitable for the lead Mark II plants.

The Applicants have taken exception to some of the acceptance criteria developed by the NRC Staff. The Staff and the Applicants are continuing to work together to resolve this matter. The Committee wishes to be kept informed.

The Mark II Owners Group and the NRC Staff are continuing to develop information relating to the method of combining loads on the containment structure. However, the Applicants have indicated that they will accept the NRC Staff's current, perhaps overly conservative, methodology, to expedite the licensing action. The Committee considers this acceptable.

The NRC Staff has determined that the present Emergency Core Cooling System analysis contains adequate margins for assessing the performance of the Zimmer Plant. It should be noted that recent tests in the Two Loop Test Apparatus (TLTA) have produced new data on the rate of vaporization of emergency core cooling water. The NRC Staff believes that further analysis of the TLTA test results may require changes in the General Electric model for calculation of this vaporization rate in order to reflect more accurately the observed physical phenomena. The Committee wishes to be kept informed.

In view of the important role of the Operational Review Committee in providing continuing reviews, and in updating and implementing safety measures, the ACRS recommends that the Operational Review Committee include additional experienced personnel from outside the corporate structure as voting members for the first few years of operation.

With regard to the generic items cited in the Committee's report, "Status of Generic Items Relating to Light Water Reactors: Report No. 6," dated November 15, 1977, those items considered relevant to Zimmer are: II-4, 5b, 6, 7, 8, 10; IIA-4; IIB-4; IIC-1, 3A, 3B, 5; IID-2. These items should be dealt with by the NRC Staff and the Applicants as solutions are found.

The Advisory Committee on Reactor Safeguards believes that, if due consideration is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, the William H. Zimmer Nuclear Power Station, Unit 1 can be operated without undue risk to the health and safety of the public.

Sincerely,

Max W Carbon

Max W. Carbon
Chairman

Honorable Joseph M. Hendrie - 3 - March 13, 1979

References:

1. Cincinnati Gas and Electric Company, "Final Safety Analysis Report, William H. Zimmer Nuclear Power Station, Unit 1," with Amendments 23 through 82.

2. U. S. Nuclear Regulatory Commission (USNRC), "Safety Evaluation Report Related to the Operation of William H. Zimmer Nuclear Power Station, Unit 1, Docket No. 50-358," USNRC Report NUREG-0528, dated January 31, 1979.

3. U. S. Nuclear Regulatory Commission, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," USNRC Report NUREG-0487, dated October, 1978.

APPENDIX C

NUCLEAR REGULATORY COMMISSION UNRESOLVED SAFETY ISSUES

C.1 Unresolved Safety Issues

The NRC staff continuously evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes from a variety of sources including experience from operating reactors, research results, NRC staff and Advisory Committee on Reactor Safeguards safety reviews, and vendor, architect/engineer and utility design reviews. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to assure safe operation is assessed. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken to assure safety, e.g., the derating of boiling water reactors as a result of the channel box wear problems in 1975. In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue prior to making licensing decisions. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing NRC staff requirements should be modified to address the issue for new plants or if backfitting is appropriate for the long-term operation of plants already under construction or in operation.

These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. Certain of these issues have been designated as "unresolved safety issues." (NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," dated January 1, 1978.) However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer term generic review is underway.

C.2 ALAB-444 Requirements

These longer term generic studies were the subject of a decision by the Atomic Safety and Licensing Appeal Board of the Nuclear Regulatory Commission. The decision was issued on November 23, 1977 (ALAB-444) in connection with the Appeal Board's consideration of the Gulf States Utility Company application for the River Bend Station, Unit Nos. 1 and 2.

In the view of the Appeal Board, (pp. 25-29):

"The responsibilities of a licensing board in the radiological health and safety sphere are not confined to the consideration and disposition of those issues which may have been presented to it by a party or an 'Interested State' with the required degree of specificity. To the contrary, irrespective of what matters may or may not have been properly placed in controversy, prior to authorizing the issuance of a construction permit the board must make the finding, inter alia, that there is 'reasonable assurance' that 'the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.' Of necessity, this 10 CFR 50.35(a) determination will entail an inquiry into whether the staff review satisfactorily has come to grips with any unresolved generic safety problems which might have an impact upon operation of the nuclear facility under consideration."

"The SER is, of course, the principal document before the licensing board which reflects the content and outcome of the staff's safety review. The board should therefore be able to look to that document to ascertain the extent to which generic unresolved safety problems which have been previously identified in a FSAR item, a Task Action Plan, an ACRS report or elsewhere have been factored into the staff's analysis for the particular reactor -- and with what result. To this end, in our view, each SER should contain a summary description of generic problems under continuing study which have both relevance to facilities of the type under review and potentially significant public safety implications."

"This summary description should include information of the kind now contained in most Task Action Plans. More specifically, there should be an indication of the investigative program which has been or will be undertaken with regard to the problem, the program's anticipated time-span, whether (and if so, what) interim measures have been devised for dealing with the problem pending the completion of the investigation, and what alternative courses of action might be available should the program not produce the envisaged result."

"In short, the board (and the public as well) should be in a position to ascertain from the SER itself -- without the need to resort to extrinsic documents -- the staff's perception of the nature and extent of the relationship between each significant unresolved generic safety question and the eventual operation of the reactor under scrutiny. Once again, this assessment might well have a direct bearing upon the ability of the licensing board to make the safety findings required of it on the construction permit level even though the generic answer to the question remains in the offing. Among other things, the furnished information likely shed light on such alternatively important considerations as whether: (1) the problem has already been resolved for the reactor under study; (2) there is a reasonable basis for concluding that a satisfactory solution will be obtained before the reactor is put in operation; or (3) the problem would have no safety implications until after several years of reactor operation and, should it not be resolved by then, alternative means will be available to insure that continued operation (if permitted at all) would not pose an undue risk to the public."

This appendix is specifically included to respond to the decision of the Atomic Safety and Licensing Appeal Board as enunciated in ALAB-444 and as applied to an operating license proceeding involving Virginia Electric and Power Company (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, NRC 245 (1978).

C.3 "Unresolved Safety Issues"

In a related matter, as a result of Congressional action on the Nuclear Regulatory Commission budget for fiscal year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977 to include, among other things, a new Section 210 as follows:

"UNRESOLVED SAFETY ISSUES PLAN"

"SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter."

The Joint Explanatory Statement of the House-Senate Conference Committee for the FY 1978 Appropriations Bill (Bill S.1131) provided the following additional information regarding the Committee's deliberations on this portion of the bill:

"SECTION 3 - UNRESOLVED SAFETY ISSUES"

"The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978. The conferees also expressed the intent that this plan should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: (1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned."

In response to the reporting requirements of the new Section 210, the NRC staff submitted to Congress on January 1, 1978, a report describing the NRC generic issues program (NUREG-0410).[17] The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the "Unresolved Safety Issues Plan" required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission indicated that "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items.

It is the NRC's view that the intent of Section 210 was to assure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of over 130 generic issues addressed in the NRC program to determine which issues fit this description and qualify as "Unresolved Safety Issues" for reporting to the Congress. The NRC review included the development of proposals by the NRC staff and review and final approval by the NRC Commissioners.

This review is described in a report, NUREG-0510, entitled "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress" dated January 1979. The report provides the following definition of an "Unresolved Safety Issue."

> "An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which the final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants it affects."

Further, the report indicates that in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety, or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an "Unresolved Safety Issue" is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, 17 "Unresolved Safety Issues" addressed by 22 tasks in the NRC program were identified. The issues are listed below. Progress on these issues was first discussed in the 1978 NRC Annual Report. The number(s) of the generic task(s) (e.g., A-1) in the NRC program addressing each issue is indicated in parentheses following the title.

"UNRESOLVED SAFETY ISSUES" (APPLICABLE TASK NOS.)

1. Water Hammer - (A-1)
2. Asymmetric Blowdown Loads on the Reactor Coolant System - (A-2)
3. Pressurized Water Reactor Steam Generator Tube Integrity -(A-3, A-4, A-5)
4. BWR Mark I and Mark II Pressure Suppression Containments - (A-6, A-7, A-8, A-39)
5. Anticipated Transients Without Scram - (A-9)
6. BWR Nozzle Cracking - (A-10)
7. Reactor Vessel Materials Toughness - (A-11)
8. Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports - (A-12)
9. Systems Interaction in Nuclear Power Plants - (A-17)
10. Environmental Qualification of Safety-Related Electrical Equipment - (A-24)
11. Reactor Vessel Pressure Transient Protection - (A-26)
12. Residual Heat Removal Requirements - (A-31)

13. Control of Heavy Loads Near Spent Fuel - (A-36)
14. Seismic Design Criteria - (A-40)
15. Pipe Cracks at Boiling Water Reactors - (A-42)
16. Containment Emergency Sump Reliability - (A-43)
17. Station Blackout - (A-44)

In the view of the staff, the "Unresolved Safety Issues" listed above are the substantive safety issues referred to by the Appeal Board in ALAB-444 when it spoke of "...those generic problems under continuing study which have...potentially significant public safety implications" (page 27). Six of the 22 tasks identified with the "Unresolved Safety Issues" are not applicable to Zimmer Unit 1 because they apply to pressurized water reactors only. These tasks are A-2, A-3, A-4, A-5, A-12, and A-26. Also, tasks A-6 and A-7 only apply to Mark I boiling water reactor containments. With regard to the 14 remaining tasks that are applicable to Zimmer Unit 1, the NRC staff has issued NUREG reports providing its proposed resolution of seven of the issues. The table below lists those issues.

Task No.	NUREG Report and Title	SER/SER Suppl. Section
A-8	NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria." October 1978. Supplement 1 to NUREG-0487, October 1980. Supplement 2 to NUREG-0487, February 1981.	6.2.
A-10	NUREG-0619 "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking"	5.2
A-24	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment."	3.11
A-31	SRP 5.4.7 and BTP 5-1, "Residual Heat Removal Systems" incorporate requirements of USI A-31.	5.4
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"	9.1
A-39	NUREG-0487 and Supplement 1 to NUREG-0487 (See above).	6.2
A-42	NUREG-0313, Rev. 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping"	5.2

The remaining issues applicable to Zimmer Unit 1 are listed in the following table.

APPENDIX C (Continued)

GENERIC TASKS ADDRESSING UNRESOLVED SAFETY ISSUES
THAT ARE APPLICABLE TO THE ZIMMER NUCLEAR STATION, UNIT 1

1.	A-1	Water Hammer
2.	A-9	ATWS
3.	A-11	Reactor Vessel Materials Toughness
4.	A-17	Systems Interaction in Nuclear Power Plants
5.	A-40	Seismic Design Criteria
6.	A-43	Containment Emergency Sump Reliability
7.	A-44	Station Blackout

With the exception of Tasks A-9, A-43 and A-44, Task Action Plans for the generic tasks above are included in NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants." A technical resolution for Task A-9 has been proposed by the NRC staff in Volume 4 of NUREG-0460, issued for comment. This served as a basis for the staff's proposal for rulemaking on this issue. The Task Action Plan for Task A-43 was issued on January 1981, and the Task Action Plan for A-44 was issued in July 1980. The information provided in NUREG-0649 meets most of the informational requirements of ALAB-444. Each Task Action Plan provides a description of the problem; the staff's appr[illegible]s to its resolution; a general discussion of the bases upon which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the manpower required; a description of the interactions with other NRC offices, the Advisory Committee on Reactor Safeguards and outside organizations; estimates of funding required for contractor supplied technical assistance; prospective dates for completing the task; and a description of potential problems that could alter the planned approach or schedule. In addition to the Task Action Plans, the staff issues, NUREG-0606, "Office of Nuclear Regulation Unresolved Safety Issues Summary, Aqua Book" on a quarterly basis which provides current schedule information for each of the Task Action Plans.

We have reviewed the 7 "Unresolved Safety Issues" listed above and the four new USIs discussed in Section C.4 as they relate to Zimmer Unit 1. Discussion of each of these issues including references to related discussions in the Safety Evaluation Report is provided below in Section C.5. Based on our review of these items, we have concluded, for the reasons set forth in Section C.5, that there is reasonable assurance that the Zimmer Nuclear Station Unit 1 can be operated prior to the ultimate resolution of these generic issues without endangering the health and safety of the public.

C.4 New "Unresolved Safety Issues"

An in-depth and systematic review of generic safety concerns identified since January 1979 has been performed by the staff, to determine if any of these issues should be designated as new "Unresolved Safety Issues." The candidate issues originated from concerns identified in NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident," ACRS recommendations, abnormal occurrence reports and other operating experience. The staff's proposed list was reviewed and commented on by the ACRS, the Office of Analysis and Evaluation of Operational Data (AEOD) and the Office of Policy Evaluation. The ACRS and AEOD also proposed that several additional "Unresolved Safety Issues" be considered by the Commission. The Commission considered the above information and approved the following four new "Unresolved Safety Issues:"

A-45 Shutdown Decay Heat Removal Requirements

A-46 Seismic Qualification of Equipment in Operating Plants

A-47 Safety Implication of Control Systems

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

A description of the above process together with a list of the issues considered is presented in NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants, Special Report to Congress," dated March 1981. An expanded discussion of each of the new "Unresolved Safety Issues" is also contained in NUREG-0705.

The applicability and bases for licensing prior to ultimate resolution of the four new USIs for Zimmer are also discussed in Section C.5.

C.5 Discussion of Tasks as they Relate to Zimmer Unit 1

A-1 Water Hammer

Water hammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions such as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Since 1971, over 200 incidents involving waterhammer in pressurized boiling water reactors have been reported. The waterhammers (or steam hammers) have involved steam generator feedrings and piping, the residual heat removal system, emergency core cooling systems, and containment spray, service water, feedwater and steam lines. Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, several waterhammer incidents have resulted in piping and valve damage. The most serious waterhammer events have occurred in the steam generator feedrings of pressurized water reactors. In no case has any waterhammer incident resulted in the release of radioactive material.

Under Generic Task A-1, the potential for waterhammer in various systems is being evaluated and appropriate requirements and systematic review procedures are being developed to ensure that waterhammer is given appropriate consideration in all areas of licensing review. A technical report, NUREG-0582, "Water Hammer in Nuclear Power Plants" (July 1979), providing the results of an NRC staff review of waterhammer events in nuclear power plants and stating current staff licensing positions, completes a major subtask of Generic Task A-1.

Although waterhammer can occur in any LWR and approximately 118 actual and probable events have been reported in BWRs as of September 1979, none have caused major pipe failures in a BWR such as Zimmer and none have resulted in the offsite release of radioactivity.

Zimmer has installed a system to preclude waterhammer from occurring in ECCS lines. This system consists of jockey pumps to keep ECCS lines water-filled so that ECCS pumps will not start pumping into voided lines and steam will not collect in the ECCS piping. To ensure that the ECCS lines remain water-filled, vents have been installed and a Tech Spec requirement to periodically vent air from the lines has been imposed. (NUREG-0528, subsection 6.3.2.).

With regard to additional protection against potential waterhammer events currently provided in plants, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) increasing valve closure times, (2) piping layout to preclude water slugs in steam lines and vapor formation in water lines, (3) use of snubbers and pipe hangers, and (4) use of vents and drains. In addition, we require that applicants conduct a preoperational vibration dynamic effects test program in accordance with Section III of the ASME Code for all ASME Class 1 and Class 2 piping systems and piping restraints during startup and initial operation. These tests will provide adequate assurance that the piping and piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips and other operating modes associated with the design operational transients.

Nonetheless, in the unlikely event that a large pipe break did result from a severe waterhammer event, core cooling is assured by the emergency core cooling systems and protection against the dynamic effects of such pipe breaks inside and outside of containment is provided.

In the event that Task A-1 identifies potentially significant waterhammer scenarios that have not explictly been accounted for in the design and operation of the Zimmer Unit, corrective measures will be required at that time. The task has not identified the need for measures beyond those already implemented.

Based on the foregoing, we have concluded that Zimmer Unit 1 can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

A-9 Anticipated Transients Without Scram (ATWS)

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as desired, then an "anticipated transient without scram," or ATWS, would have occurred.

All BWRs including Zimmer have been required to provide recirculation pump trip in the event of a reactor trip and to provide additional operator training for recovery from ATWS events.

A recirculation pump trip provision has been incorporated in the Zimmer design. In addition, emergency procedures and operator training have been implemented to cope with potential ATWS events (Subsection 15.2 of this supplement to NUREG-0528).

Operator training and action as described, in conjunction with the automatic recirculation pump trip, significantly improves the automatic recirculation pump trip, significantly improves the capability of the facility to withstand a range of ATWS events, such that operation of this facility presents no undue risk to the health and safety of the public while this matter is under review.

That ATWS issue is currently scheduled for rulemaking in mid-summer 1981. The applicant will be required to comply with any further requirements on ATWS which may be imposed as a result of the rulemaking.

Based on our review, we have concluded that there is reasonable assurance that Zimmer can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-11 Reactor Vessel Materials Toughness

Resistence to brittle fracture is described quantitatively by a material property generally denoted as "fracture toughness." Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in nuclear reactor pressure vessels, three considerations are important. First, fracture toughness increases with increasing temperature. Second, fracture toughness decreases with increasing load rates. Third, fracture toughness decreases with neutron irradiation.

In recognition of these conditions, power reactors are operated within restrictions imposed by the Technical Specifications on pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel material. The effect of neutron radiation on fracture toughness of the vessel material over the life of the plant is accounted for in Technical Specification limitations.

The principal objective of Task A-11 is to develop safety criteria to allow a more precise assessment of safety margins during normal operation, transients and accident conditions in older reactor vessels with marginal fracture toughness.

The staff's review of the fracture toughness of the Zimmer reactor vessel is incomplete at this time. However, based upon our evaluation to date of the Zimmer reactor vessel materials toughness, we have concluded that this unit will have adequate safety margins for initial operation against brittle failure during operating, testing, maintenance, and anticipated transient conditions. When Task Action Plan A-11 is completed and explicit fracture evaluation criteria for accident conditions are defined, all vessels will be reevaluated for acceptability over their design lives. Since Task A-11 is projected to be completed well in advance of the Zimmer reactor vessel reacting a level of marginal fracture resistance, acceptable vessel integrity for the postulated accident conditions will be assured at least until the reactor vessel is reevaluated for long-term acceptability. The staff's evaluation for long-term acceptability will be provided in a supplement to the Safety Evaluation Report (NUREG-0528, Supplement 1, Subsection 3.5).

Therefore, based upon the foregoing, we have concluded that Zimmer can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

A-17 Systems Interaction in Nuclear Power Plants

The licensing requirements and procedures used in our safety review address many different types of systems interactions. Current licensing requirements are founded on the principle of defense-in-depth. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems, and protection against events such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage. These design provisions supplemented by the current review procedures of the Standard Review Plan (NUREG-75/087) which require interdisciplinary reviews and which account, to a large extent, for review of potential systems interactions, provide for an adequately safe situation with respect to such interactions, provide for an adequately safe situation with respect to such interactions. The quality assurance program which is followed during the design, construction, and operational phases for each plant is expected to provide added assurance against the potential for adverse systems interactions.

In November 1974, the Advisory Committee on Reactor Safeguards requested that the NRC staff give attention to the evaluation of safety systems from a multidisciplinary point of view, in order to identify potentially undesirable interactions between plant systems. The concern arises because the design and analysis of systems is frequently assigned to teams with functional engineering specialties--such as civil, electrical, mechanical, or nuclear. The question is whether the work of these functional specialists is sufficiently integrated in their design and analysis activities to enable them to identify adverse interactions between and among systems. Such adverse events might occur, for example, because designers did not assure that redundancy and independence of safety systems were provided under all conditions of operation required, which might happen if the functional teams were not adequately coordinated.

In mid-1977, Task A-17 was initiated to confirm that present review procedures and safety criteria provide an acceptable level of redundancy and independence for systems required for safety by evaluating the potential for undesirable interactions between and among systems.

The NRC staff's current review procedures assign primary responsibility for review of various technical areas and safety systems to specific organization units and assign secondary responsibility to other units where there is a function or interdisciplinary relationship. Designers follow somewhat similar procedures and provide for interdisciplinary reviews and analyses of systems. Task A-17 will provide an independent study of methods that could identify important systems interactions adversely impacting safety; and which are not considered by current review procedures. The first phase of this study began in May 1978, and was completed in February 1980 by Sandia Laboratories under contract to the NRC staff.

The Phase 1 investigation was structured to identify areas where interactions are possible between and among systems and have the potential of negating or seriously degrading the performance of safety functions. The study concentrated on common cause or linking failures among systems that could violate a safety function. The investigation then identified where NRC review procedures may not have properly accounted for these interactions.

The Sandia Study used fault-tree methods to identify component failure combinations (cut-sets) that could result in loss of a safety function. The cut-sets were reduced to minimal combinations by incorporating six common or linking systems failures into the analysis. The results of the Phase 1 effort indicate that, within the scope of the study, only a few areas of review procedures need improvement regarding systems interaction. However, the level of detail needed t identify all examples of potential system interaction candidates observe n some operating plants was not within the Phase 1 scope of the Sandia Study.

The Systems Interaction Branch formed in NRR in April 1980, has been studying state-of-the-art methods that can be used to predict systems interactions. The initial effort, supported by three laboratory contractors, is underway; a range of methods is being considered and tested for feasibility against a sample of some systems interaction candidates derived from Licensee Event Report evaluations

It is expected that the development of systematic ways to identify and evaluate systems interactions will reduce the likelihood of common cause failures resulting in the loss of plant safety functions. However, the studies to date indicate that current review procedures and criteria supplemented by the application of post-TMI findings and risk studies provide reasonable assurance that the effects of potential systems interaction on plant safety will be within the effects on plant safety previously evaluated.

Therefore, we concluded that there is reasonable assurance that Zimmer Unit 1 can be operated prior to the final resolution of this generic issue without endangering the health and safety of the public.

A-40 Seismic Design Criteria - Short-Term Program

NRC regulations require that nuclear power plant structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in the NRC regulations and in Regulatory Guides issued by the Commission. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, rereviews of the seismic design of various plants are being undertaken to assure that these plants do not present an undue risk to the public. Task A-40 is, in effect, a compendium of short-term efforts to support the reevaluation efforts of the NRC staff, especially those related to older operating plants. In addition, some revisions to SRP sections and regulatory guides to bring them more in line with the state-of-the-art will result.

The seismic design basis and seismic design of Zimmer Unit 1 has been evaluated at the operating license stage and have been found acceptable. Seismic design review of Zimmer was conducted using current licensing criteria and requirements (Subsection 3.7, 3.8 and 3.9 of NUREG-0528). Should the resolution of Task A-40 indicate a change is needed in these licensing requirements, all operating reactors, including Zimmer will be reevaluated on a case-by-case basis. Accordingly, we have concluded that Zimmer Unit 1 can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

A-43 Containment Emergency Sump Reliability

Following a postulated loss-of-coolant accident, i.e., a break in the reactor coolant system piping, the water flowing from the break would be collected in the suppression pool. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water would also be circulated through the containment spray system to remove heat and fission products from the drywell and wetwell atmosphere. Loss of the ability to draw water from the suppression pool could disable the emergency core cooling and containment spray systems.

The concern addressed by this Task Action Plan for boiling water reactors (BWRs) is limited to the potential for degraded ECCS performance as a result of thermal insulation debris that may be blown into the suppression pool during a loss-of-coolant accident and cause blockage of the pump suction lines. A second concern, potential vortex formation, is not considered a serious concern for Mark II containment due to the large depth of the pool (approximately 25 feet) and the low approach velocities (NUREG-0528, Subsection 6.2.2).

With regard to potential blockage of the intake lines, the likelihood of any insulation being drawn into an ECCS pump suction line is very small. The potential debris in the drywell could only be swept into the suppression pool via the downcomer piping. However, the downcomer pipes (2-foot diameter) are capped with jet deflectors and would prevent any large pieces from reaching the suppression pool. Any smaller pieces reaching the pool would tend to settle on the bottom and would not be drawn into the pump suction since it is located several feet above the pool bottom. In addition, BWR designs employ strainers within the suction piping and NPSH calculations for the pumps are based on an assumed blockage of 50%.

Accordingly, we have concluded that Zimmer can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public (NUREG-0528, Subsection 6.2.2).

A-44 Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes an offsite alternating current (ac) power connection, a standby emergency diesel generator ac power supply, and direct current (dc) sources.

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all ac power, i.e., a loss of both the offsite and the emergency diesel generator ac power supplies. This issue arose because of operating experience regarding the reliability of ac power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. During each of these loss of offsite power events, the onsite emergency ac power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In addition, there have been numerous reports of emergency diesel generators failing to start and run in operating plants during periodic surveillance tests.

A loss of all ac power was not a design basis event for the Zimmer facility. Nonetheless, a combination of design, operation and testing requirements that have been imposed on the applicant will assure that these units will have substantial resistance to a loss of all ac and that, even if a loss of all ac should occur, there is reasonable assurance that the core will be cooled. These are discussed below.

A loss of offsite ac power involves a loss of both the preferred and backup sources of offsite power. Our review and basis for acceptance of the design, inspection, and testing provisions for the offsite power system are described in Subsection 8.1 of this supplement to NUREG-0528.

If offsite alternating current power is lost, three diesel-generators and their associated distribution systems will deliver emergency power to safety-related equipment. Our review of the design, testing, surveillance, and maintenance provisions for the onsite emergency diesels is described in Subsection 8.1 of this supplement to NUREG-0528. Our requirements include preoperational testing to assure the reliability of the installed diesel-generators in accordance with our requirements discussed in this report. In addition, the applicant has been requested to implement a program for enhancement of diesel-generator reliability to better assure the long-term reliability of the diesel-generators. This program resulted from the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Generator Reliability."

If both offsite and onsite ac power are lost, boiling water reactors may use a combination of safety/relief valves and the reactor core isolation cooling system (RCIC) to remove decay heat without reliance on ac power. These systems assure that adequate cooling can be maintained for at least two hours, which allows time for restoration of ac power from either offsite or onsite sources.

The issue of station blackout was also considered by the Atomic Safety and Licensing Appeal Board (ALAB-603) for the St. Lucie Unit No. 2 facility. In addition, in view of the completion schedule for Task A-44 (October 1982), the Appeal Board recommended that the Commission take expeditious action to ensure that other plants and their operators are equipped to accommodate a station blackout event. The Commission has reviewed this recommendation and determined that some interim measures should be taken at all facilities including Zimmer while Task A-44 is being conducted. Consequently, interim emergency procedures and operator training for safe operation of the facility and restoration of ac power will be required. The staff notified the applicant of these requirements in a letter from D Eisenhut, NRC, to the applicant dated February 25, 1981. We will condition the operating license for Zimmer that these procedures and this training be completed by fuel load date.

Based on the above, we have concluded that there is reasonable assurance that Zimmer Unit No. 1 can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-45 Shutdown Decay Heat Removal Requirements

Following a reactor shutdown, the radioactive decay of fission products continues to produce heat (decay heat) which must be removed from the primary system. The principal means for removing this heat in a boiling water reactor while at high pressure is via the steam lines to the turbine condenser. The condensate is normally returned to the turbine condenser. The condensate is normally returned to the reactor vessel by the feedwater system; however, the steam turbine driven reactor core isolation cooling system (RCIC) is provided to maintain primary system inventory, if ac power is not available. When the system is at low pressure, the decay heat is removed by the residual heat removal systems (RHR). This USI will evaluate the benefit of providing alternate means of decay heat removal which could substantially increase the plants' capability to handle a broader spectrum of transients and accidents. The study will consist of a generic system evaluation and will result in recommendations regarding the desirability of and possible design requirements for improvements in existing systems or an alternative decay heat removal method if the improvements or alternative can significantly reduce the overall risk to the public.

The Zimmer reactor has various methods for the removal of decay heat. As discussed above, the decay heat is normally rejected to the turbine condenser and returned to the vessel by either the feedwater system or the reactor core isolation cooling system (RCIC) (from the condensate storage tank). If the condenser is not available (e.g., loss of offsite power), heat can be removed via the safety/relief valves to the suppression pool. Also, the high pressure core spray (HPCS) system is provided if the RCIC is not available. Both of these systems can recirculate fluid to the vessel from either the condensate storage tank or the suppression pool. If the RCIC and HPCS are unavailable, the reactor system pressure can be reduced by the automatic depressurization system (ADS) so that cooling by the RHR can be initiated. When the condenser is not used, the heat rejected to the suppression pool is subsequently removed by the residual heat removal system (RHR).

The RCIC and HPCS systems at Zimmer have improvements over comparable systems at older BWRs. The RCIC has been upgraded to safety grade quality (now required for all BWRs), and the HPCS is powered by its own diesel generator so it can operate with an assumed loss of all other sources of ac power. Also, the RHR contains three pumps; the flow capacity of any single pump is sufficient to easily remove the decay heat. Accordingly, we have concluded that Zimmer can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public (Subsections 6.3, 7.3 and 7.4 of NUREG-0528 and this supplement).

A-46 Seismic Qualification of Equipment in Operating Plants

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this Unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shut down the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

Zimmer Unit 1 was designed using current seismic criteria and the design has been reviewed and approved by the Commission staff in accordance with current design criteria and methods for seismic qualification. Therefore, we conclude that Zimmer Unit 1 can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public (Subsections 3.9 and 3.10 of this supplement to NUREG-0528).

A-47 Safety Implications of Control Systems

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern is the potential for a single failure such as a loss of a power supply, short circuit, open circuit, or sensor failure to cause simultaneous malfunction of several control features. Such an occurrence would conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern is for a postulated accident to cause control system failures which would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. Although it is generally believed that such control system failures would not lead to serious events or results in conditions that safety systems cannot safely handle, in-depth studies have not been rigorously performed to verify this belief. The potential for an accident that would affect a particular control system, and effects of the control system failures, may differ from plant to plant. Therefore, it is not possible to develop generic answers to these concerns, but rather plant-specific reviews are required. The purpose of this USI is to define generic criteria that will be used for plant-specific reviews.

The Zimmer control and safety systems have been designed with the goal of ensuring that control system failures (either single or multiple failures) will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any "anticipated operational occurrence" or "accident."

This has been accomplished by either providing independence between safety and nonsafety systems or providing isolating devices between safety and nonsafety systems. These devices preclude the propagation of nonsafety system equipment faults such that operation of the safety system equipment is not impaired.

A systematic evaluation of the control system design, such as contemplated for this USI, has not been performed to determine whether postulated accidents could cause significant control system failures which would make the accident consequences more severe than presently analyzed. However, a wide range of bounding transients and accidents is presently analyzed to assure that the postulated events would be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems have been performed with the goal of ensuring that control system failures (single or multiple) will not defeat safety system action.

A specific subtask of this USI issue will be to study the reactor overfill transient in BWRs to determine the need for preventative and/or mitigating design measures to preclude or minimize the consequences of this transient. Several early BWRs have experienced reactor vessel overfill transients with subsequent two-phase or liquid flow through the safety/relief valves. Following these early events, control grade high level trips (level 8) have been installed at most BWRs (including Zimmer) to terminate flow from the appropriate systems. These high level trips are single failure proof and periodic surveillance is required by the Technical Specifications. No overfilling events have occurred since the level 8 trips were installed.

Based on the above, we have concluded that there is reasonable assurance that Zimmer Unit No. 1 can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Following a loss-of-coolant accident in a light water reactor (LWR) plant, combustible gases, principally hydrogen, may accumulate inside the primary reactor containment as a result of: (a) metal-water reaction involving the fuel element cladding; (2) the radiolytic decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

Because of the potential for significant hydrogen generation as the result of an accident, 10 CFR Section 50.44, "Standards for Combustible Gas Control System in Light-Water Cooled Power Reactors," and General Design Criterion 41, "Containment Atmosphere Cleanup," in Appendix A to 10 CFR Part 50 require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

The Regulation, 10 CFR Section 50.44, requires that the combustible gas control system provided be capable of handling the hydrogen generated as a result of degradation of the emergency core cooling system such that the hydrogen release is five times the amount calculated in demonstrating compliance with 10 CFR Section 50.46 or the amount corresponding to reaction of the cladding to a depth of 0.00023 inch, whichever amount is greater.

The accident at TMI-2 on March 28, 1979 resulted in hydrogen generation well in excess of the amounts specified in 10 CFR Section 50.44. As a result of this knowledge it became apparent to NRC that specific design measures are needed for handling larger hydrogen releases, particularly for smaller, low pressure containments. As a result, the Commission determined that a rulemaking proceeding should be undertaken to define the manner and extent to which hydrogen evolution and other effects of a degraded core need to be taken into account in plant design. An advanced notice of this rulemaking proceeding on degraded core issues was published in the Federal Register on October 2, 1980.

Recognizing that a number of years may be required to complete this rulemaking proceeding, a set of short-term or interim actions relative to hydrogen control requirements were developed and implemented. These interim measures were described in a second October 2, 1980 Federal Register notice. For plants with small containments (Mark I and Mark II) such as Zimmer, the interim rule specified that inerting is required to preclude hydrogen burning.

Zimmer has committed to inerting the containment building during power operation. We, therefore, conclude that Zimmer can be operated prior to resolution of this unresolved safety issue and the proposed rulemaking without undue risk to the health and safety of the public.

APPENDIX D

ERRATA

PAGE 1-6	Last sentence of first partial paragraph should read, "The reactor vessel is fabricated of low alloy steel and is clad internally with stainless steel except for the top head, and major nozzles such as the feed water nozzles. Bottom head nozzles are clad with inconel."
1-9	Following the last paragraph in subsection 1.2.9, add the following paragraphs: "The containment hydrogen recombiner system instrumentation and control provide for manual starting of the hydrogen recombiner used to maintain hydrogen-oxygen level within the primary containment below the flammability limit in the event of a loss-of-coolant accident." "The standby gas treatment system instrumentation and control provide for automatic and manual starting of the standby gas treatment system used to treat the reactor building gas volume prior to release to the environment and to maintain a slightly negative internal building pressure when appropriate."
1-17	Sentence in third paragraph should read ".... or not we should grant certain exemptions..."
3-2	Following the last paragraph, add the paragraph "We conclude that the fluid systems pressure-retaining components important to safety, that have been designed, fabricated, erected and tested to quality standards in conformance with the Nuclear Regulatory Commission's regulations, the applicable Regulatory Guides and industrial codes and standards, are acceptable."
3-17	From the second sentence in the first paragraph, delete "and their restraints" In the first sentence of the second paragraph replace "restraints" by "supports".
3-20	Third sentence in last paragraph should read, "On the basis....., site visit and in situ confirmatory test results for hydrodynamic loads the seismic qualification.... be retested."
4-10	In second paragraph delete "(see introduction to Section 15.0 of this report)."
5-11	First sentence in second full paragraph should read "We are reviewing ....specific exemptions should be granted...."

5-11 From first paragraph in subsection 5.3.2, delete word, "conservative".

5-12 Delete first sentence on page.

5-16 In sixth paragraph, insert "and" before "pipe" in second sentence and delete third sentence.

6-30 Add to the first full subsection the following paragraph,

"It should be noted, however, that the applicant plans to operate the reactor with the control rod drive return lines valved out (see subsection 5.2.3)."

6-44 In first sentence of subsection 6.4.1, delete "redundant" insert "a" before "once-through", change "trains" to "train" and in second sentence change "50" to "100" and delete "full". From the first sentence in second paragraph of subsection 6.4.1, delete "Each of" and change "trains" to "train".

7-9 In the fourth paragraph, delete second sentence. Third sentence should read, "The applicant did not initially agree with our position; however, the design was modified to provide for automatic transfer of the reactor core isolation cooling system suction from the condensate system to the suppression pool."

7-21 Delete "Drywell and" from first subtitle on page. In first full paragraph, replace "drywell" with "wetwell" in first sentence. Add to first full paragraph "This instrumentation is acceptable to us because it meets position C.5 of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident."

9-12 First full paragraph should read, "Each of the standby diesel generators is provided with independent compressed air starting systems consisting of two air compressors, each compressor supplying two storage tanks. Each pair of tanks is capable of providing five starts without recharging from the aircompressors. The starting air system is designed to seismic Category I requirements and meets the guidance and criteria described in the Standard Review Plan 9.5.6 "Emergency Diesel Engine Starting System." We conclude, based on our review, that the diesel generator starting systems are acceptable."

11-9 From third sentence in first paragraph of subsection 11.2.3 delete "ventilation air filters" and comma after "paper". Add, "HEPA filters will be shipped in wooden boxes in accordance with Department of Transportation packaging and shipping regulations.

11-11 Second item in Table 11-2 should read "4.2 x 10-6 microcuries per milliliter (Cs-137)" Seventh item in Table 11-2 should read "Ion Chamber".

17-2 Add "*" to Station Quality Engineer box.

17-4	In last sentence of last paragraph, delete "the Principle Quality Assurance and Standards Engineer." and replace by "reviewed by the Station Quality Engineer."
	<u>Significant Typographical Errors</u>
vi	Change "15-14" to "15-13" and "15-16" to "15-15"
1-13	Second item in Table 1-1, change "Inside" to "Outside".
1-16	In second sentence of next to last paragraph add "of" between "evaluation" and "this."
1-18	In first line, delete "7.6" and change "7.5" to "7.5.3".
3-4	In last paragraph of subsection 3.4.1, change "its" to "our".
3-38	In first sentence of fourth full paragraph, delete, "the requirements of".
4-6	In last sentence of first paragraph, change "general to "generic".
4-8	In first sentence of final paragraph, change "(N0.05-1.5 full power)" to "(N05-1.5 full power)"
4-14	In the last sentence of first full paragraph, change "<u>J</u>" to "<u>J</u>".
5-4	In middle paragraph, change "1165" to "1150".
5-7	In first sentence of fifth full paragraph, delete words "intent of".
7-6	In second line, change "buss" to "bus".
7-13	In fourth line of first full paragraph, delete period and use lower case w.
7-24	In third line of first paragraph, insert "to" between "subject" and "some".
7-30	Delete second "Reactor Manual Control System" subtitle.
11-4	In next to last sentence in third full paragraph, delete second "Cooled Nuclear".
11-8	In third line, change "present" to "preset".
15-3	Second line of last paragraph, change "header" to "heater".
15-10	In last two lines of Table 15-1, change "F" to "F" (three places).
15-14	In last line, change "WK/K" to "WK/K".

APPENDIX E

FIRE PROTECTION
SAFETY EVALUATION REPORT
BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
U.S. NUCLEAR REGULATORY COMMISSION
IN THE MATTER OF THE
CINCINNATI GAS & ELECTRIC COMPANY
WM. H. ZIMMER NUCLEAR PLANT
UNIT NO. 1
DOCKET NO. 50-358
SEPTEMBER 19, 1979

APPENDIX E

WM. H. ZIMMER EVALUATION REPORT
FIRE PROTECTION REVIEW
UNIT NO. 1

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APPENDIX E

FIRE PROTECTION REVIEW

I. INTRODUCTION

We have reviewed the Wm. H. Zimmer Fire Protection Program Reevaluation and Fire Hazards Analysis submitted by the applicant by letter dated February 18, 1977, including Revisions 1 through 14. The reevaluation was in response to our request to evaluate their fire protection program against the guidelines of Appendix A to Branch Technical Position (BTP) APSCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." As part of our review, we visited the plant site on two occasions to examine the relationship of safety related components, systems, and structures in specific plant areas to both combustible materials and to associated fire detection and suppression systems. The overall objective of our review was to ensure that in the event of a fire at Zimmer Unit No. 1, personnel and the plant equipment would be adequate to safely shutdown the reactor, to maintain the plant in a safe shutdown condition, and to minimize the release of radioactivity to the environment.

Our review included an evaluation of the automatic and manually operated water and gas fire suppression systems, the fire detection systems, fire barriers, fire doors and dampers, fire protection administrative controls, fire brigade size and training, and the plant fire protection Technical Specifications.

Our conclusion, given in Section VII is that the Fire Protection Program of the Wm. H. Zimmer Plant, with the proposed improvements meets the guidelines contained in Appendix A to BTP ASB 9.5-1. On February 19, 1981, the Commission approved a rule concerning fire protection. Although this rule and its Appendix R are not directly applicable to Wm. H. Zimmer, the requirements set forth in Appendix R are being used as guidelines in the licensing of plants after January 1, 1979. On April 27, 1981, the Commission required that operating licenses issued after January 1, 1979, contain a condition requiring compliance with commitments made by an applicant and agreed to by the staff after differences between the applicant's program and the guidelines set forth in Appendix A to BTP 9.5-1 and Appendix R to 10 CFR 50 have been identified and evaluated.

The applicant has not committed to meet the technical requirements of Appendix R to 10 CFR, Part 50, or provide equivalent protection. Since the review of Wm. H. Zimmer was not specifically conducted to Appendix R requirements, we will require the applicant to meet the technical requirements of Appendix R to 10 CFR, Part 50, or provide equivalent protection. Our consultants, Gage-Babcock and Associates, Inc., who participated in the review of the fire protection program and in the preparation of this safety evaluation report, concur with our findings.

II. FIRE PROTECTION SYSTEMS DESCRIPTION AND EVALUATION

A. Water Supply Systems

The water supply system consists of two fire pumps separately connected to a buried, 12-inch pipe loop around the plant. The fire pumps are rated 2500 gpm at 125 psig head, and one is motor driven and the other is diesel engine driven. The water supply source consists of five million gallons of water contained in the cooling tower basin, which will be kept full by two 15,000 gpm make-up pumps taking water from the Ohio River. In the event that the cooling tower basin is drained for maintenance, the water level in the pump structure and supply tunnel would exceed the required 300,000 gallons.

A separate 40 gpm pressure maintenance pump (jockey pump) maintains the system pressure at 130 psig. If the water supply system pressure falls to 125 psig then the motor driven fire pump starts. The diesel pump actuates if the pressure falls to 115 psig. The fire pumps are located in separate fire pump rooms with separate alarms provided in the control room to monitor pump operation, prime mover availability, or failure of a fire pump to start. To preclude the possibility of the collapse of the common roof, we requested and the applicant in revision 7 agreed to apply a 3-hour fire rating to the structural steel members. The power supply associated with the control signal which automatically starts the fire pumps is supplied by the Class 1E station battery system. Both fire pumps are UL listed.

The fire suppressions system requiring the greatest water demand for areas containing or exposing safety related equipment or circuits is the cable spreading room sprinkler system. This water flow requirement is 950 gpm and, coupled with 750 gpm for hose streams, creates a total water demand of 1,700 gpm. Since the system can deliver 2,500 gpm, the water supply system is adequate and is, therefore, acceptable.

B. Sprinkler and Standpipe Systems

The automatic/manual sprinkler systems and the hose stations are connected to the interior water supply header. The interior water supply system is fed from two separate supply connections to the looped yard system with appropriate valves to perform maintenance or to prevent a single break from impairing the entire distribution system. The water supply valves to the sprinklers are electrically supervised. Also, actuation of any water fire suppression system will cause a fire pump to start on a low header pressure signal. The low pressure alarm and a pump running signal indicate in the control room. Additionally, the automatic sprinkler systems have water flow alarms which indicate in the control room.

The automatic sprinkler systems, e.g., wet pipe sprinkler system, preaction sprinkler systems, deluge, and water spray systems, are designed to the requirements of National Fire Protection Association (NFPA) Standard No. 13, "Standard for Installation of Sprinkler Systems," and NFPA Standard No. 15, "Standard for Water Spray Fixed System."

Manual hose stations are located throughout the plant to ensure that an effective hose stream can be directed to any safety related area in the plant. These systems are consistent with the requirements of NFPA Standard No. 14, "Standpipe and Hose Systems" for sizing, spacing, and pipe support requirements.

The areas that have been equipped with automatic water suppression systems are as follows:

Deluge Systems

Turbine Bldg. (portions of)
HVAC and Lab Area (Purge Filter)
HVAC Area (El. 567') Filter
Reactor Bldg. Gas Treatment Filters

Wet Pipe

Turbine Bldg. (portions of)
Radwaste Bldg.
Fork Lift Truck Route*
Off-Gas Area*
Reactor Building*
Heater Bay*
HVAC Area*
Various Manholes*
Diesel Day Tank Cubicles
Diesel Fire Pump Room

Pre-Action

Cable Spreading Room

We have reviewed the design criteria and the bases for the water suppression systems. We conclude that these systems meet the guidelines of Appendix A to BTP ASB 9.5-1 and are, therefore, acceptable.

C. Gas Fire Suppression Systems

Total flooding CO_2 systems are provided for the Diesel Generator Rooms and they are actuated by heat detection systems. The CO_2 system is designed to discharge, after a 60-second delay, a 34% concentration. The Computer Room is protected by a Halon 1301 gas system activated by a cross-zoned smoke detection system and is designed to provide a 5% Halon concentration.

We have reviewed the design criteria and bases for the CO_2 and Halon fire suppression systems. We conclude that these systems satisfy the provisions of Appendix A to BTP ASB 9.5-1 and are in accordance with the applicable portions of the NFPA codes, and are, therefore, acceptable.

D. Fire Detection Systems

The fire detection systems consist of the detectors, associated electrical power supplies, and the annunciation panels. The types of detectors used

*Sprinkler system installed at our request.

at the Zimmer Nuclear Plant are ionization (products of combustion) and thermal (heat sensors). Fire detection systems give an audible and visual alarm which annunciates in the plant control room. Local audible and/or visual alarms are also provided. Both types of fire detection systems are connected to the emergency power supply.

The fire detection systems have been or will be installed according to NFPA No. 72D, "Standard for the Installation, Maintenance, and Use of Proprietary Protection Signalling Systems."

We have reviewed the fire detection systems to ensure that fire detectors are adequate to provide detection and alarm of fires that could occur. We have also reviewed the fire detection system's design criteria to ensure that they conform to the applicable sections of NFPA No. 72D. We conclude that the design and the installation of the fire detection systems meet the guidelines of Appendix A to BTP ASB 9.5-1 and are, therefore, acceptable.

III. OTHER ITEMS RELATED TO THE STATION FIRE PROTECTION PROGRAM

A. Fire Barriers and Fire Barrier Penetrations

All structural steel members in areas comprising the control room, switchgear rooms, cable spreading room and the auxiliary equipment room are protected by a 3-hour rated fire resistant covering, which meets our separation criteria. The separating walls between the control room, switchgear rooms, cable spreading room, auxiliary equipment room, reactor building, turbine building and auxiliary building are a minimum 24-inch-thick structural reinforced concerete walls or minimum 12-inch-thick solid concrete masonry unit walls and carry a 3-hour fire rating.

The floor/ceiling assemblies for the control room (floor only), cable spreading room, and the auxiliary equipment room (roof only) are rated for 1-1/2 hours. We have evaluated the fuel loading, fire detection and suppression, and the effects of fire breaching the barrier for areas having a construction assembly with a fire rating of 1-1/2 hours. Based on our evaluation, we have concluded that the 1-1/2 hour barriers are acceptable in these areas. Other areas of the plant not listed above have appropriate and acceptable fire barriers.

The applicant has provided acceptable documentation, viz., reference to specific UL designs, to substantiate the fire rating of the barriers and 3-hour penetration seals used in the penetrations for cable trays, conduits, and piping. We have concluded that the fire barrier ratings meet the guidelines of Appendix A to BTP 9.5-1, and, therefore, are acceptable.

B. Fire Doors and Dampers

We have reviewed the placement of fire doors to ensure that fire doors of the proper fire rating have been provided.

The licensee has provided 3-hour ventilation fire dampers for 3-hour wall, ceiling/ floor assemblies.

The fire barriers, barrier penetrations, fire doors and dampers will be provided in accordance with the guidelines of Appendix A to BTP ASB 9.5-1 and, therefore, are acceptable.

IV. ALTERNATE SHUTDOWN

The evaluation of the alternate shutdown capability is reported in NUREG-0528, Chapter 7, page 7-23, dated January, 1979, and found to be acceptable.

Because of our concern for the affects of a single fire event in the control room or the cable spreading room, we requested and the applicant agreed to provide an alternate shutdown capability, documented in revision 13, to allow the plant to be brought to a cold shutdown independent of the control room and cable spreading room. Two redundant shutdown panels are provided, each separated and located in separate switchgear rooms. A fire in either the control or spreading room would not jeopardize operation of the alternate shutdown panels nor would a fire in either of the panels cause malfunctions in the control room or the cable spreading room. Division I cable enters the control room from above, whereas, Division II enters from the cable spreading room below. Only two cable risers of Division I are in the cable spreading room, but these risers are enclosed with a 1-1/2 hour fire rated barrier; hence, in effect, fully separated from the cable spreading room. A single fire event cannot impair safe shutdown because of the fire protection provided and the presence of two independent shutdown panels.

V. FIRE PROTECTION FOR SPECIFIC AREAS

A. Cable Spreading Room

The cable spreading room is separated from the balance of the plant by 3-hour rated fire walls and a 1-1/2 hour fire rated ceiling and floor assembly. These barriers are discussed in Section III and found to be acceptable. Two access doors to the spreading room are located from a common corridor to provide access from two directions.

An automatic, pre-action sprinkler, with water spray on each cable tray, will be installed in the cable spreading room before fuel loading. The sprinkler system serves as the primary fire extinguishing system. Additional backup is provided by standpipe systems and portable extinguishers. Portable fans are available for smoke venting. In addition, installed smoke detectors will initiate an early warning alarm in the control room prior to the sprinkler system actuation.

We were initially concerned that a fire could affect redundant shutdown systems in the cable spreading room. However, in revision 8 the applicant committed to install a 1-1/2 hour fire rated barrier for the two Division 1 cable risers located in this room. Further, as discussed in Section IV, the applicant has installed redundant emergency shutdown panels so that alternate shutdown capability exists independent of the cable spreading room. The fire protection for the cable spreading room meets the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore, acceptable.

B. Reactor Building and Containment

A lube oil fire hazard was initially thought to be associated with the Primary Recirculation Pumps (PRP) located in the primary containment. The pump is lubricated and cooled by water, but the pump motor contains lube oil. The pump motor lube oil systems is contained within a metal motor housing with no external parts such as piping, flanges, valves, and coolers. Hence, an engineered oil leak collection system or additional fire protection for the pumps is not required.

The essential divisional cable penetrations are separated into opposite quadrants of the containment and reactor building. The HPCS, LPCS, RHR, and RCIC equipment is separated and, also, located in opposite reactor building quadrants. Further, these cooling systems are isolated by 3-hour fire rated barriers. No fire hazards are nearby nor would any transient exposure fire threaten any two divisions of equipment simultaneously. Any redundant cable divisions routed in close proximity to each other will be protected as indicated in Section V.E. prior to fuel loading.

The Reactor Building firewater supply is directly connected to the yard fire main system. Other building fire protection features include 3-hour fire rated barriers, standpipes, local deluge systems, and fire extinguishers.

We have reviewed the applicant's Fire Hazards Analysis for the areas inside the containment and reactor building, and conclude that with the proposed modifications the fire protection will meet the guidelines of Appendix A to BTP ASB 9.5-1 and is, therefore acceptable.

C. Fork Lift Truck Route (El. 525')

We were concerned with the Fork Lift Truck Route Fire Area (El. 525') because of the presence of all three electrical divisions in close proximity to each other. This area is about 1,600 square feet and is enclosed by 3-hour fire rated walls, floors, and ceiling except for a large opening into the turbine building where a water curtain will be installed. The area is protected by automatic wet pipe sprinkler systems based on a design density of 0.30 gpm/ft2. Two levels of sprinklers are provided, one at the ceiling level and one below the ducts, cable trays, and other obstructions. Each system is on a separate connection from the auxiliary building water supply header. Should a fire occur in this area and the sprinkler system fails, then all three divisions could be affected. The applicant, at our request, has proposed in revisions 7, 11, 13, and 14, to isolate and totally enclose one of the redundant safety divisions with a 2-hour fire rated barrier. The barrier design will conform to an ASTM E-119 fire tested design identified as OSU T-4410. Hence, in effect, the enclosed space with its cabling, will not be considered part of the Fort Lift Truck Route fire area, but will now be incorporated as being part of the diesel generating room where this safety division originates. Hence, two independent and separated plant areas will result and redundant safety related divisions of cable will not be exposed to any one single fire event.

We have reviewed the Fork Lift Truck Route area and conclude that with the proposed modifications the fire protection meets the guidelines of Appendix A, BTP 9.5-1 and is, therefore, acceptable.

D. Auxiliary Equipment Room

The auxiliary equipment room contains all three redundant cable divisions and electrical panels. The applicant, at our request, in revision 13, agreed to install a 1-1/2 hour fire rated wall to separate this area into two separate rooms. The wall, in conjunction with the cable protection specified for this area, will eliminate the concern of having required redundant safety divisions necessary for safe shutdown in close proximity to each other.

We have reviewed the auxiliary equipment room and conclude that the wall modification will result in separate rooms, and thus meets the guidelines of Appendix A to BTP 9.5-1 and is, therefore, acceptable.

E. Plant Areas Containing Redundant Divisions

A number of plant areas have physical arrangements wherein redundant divisions of cables/conduits and equipment are in close proximity to each other and, therefore, could be vulnerable to a single, transient fire event. Originally, the applicant was relying solely on administrative controls to preclude a fire event from taking place in affected areas. Based on experience, administrative controls alone are not sufficient to prevent storage of combustibles, occurrence of all ignition sources, etc. We requested, and the applicant agreed in revision 13, to provide 1-1/2 hour fire rated barriers for one of the divisions where it was agreed that no automatic fire suppression will be installed. Other areas will be provided with an automatic sprinkler system and a minimum 1/2-hour fire rated barrier enclosing each redundant cable system. This additional measure will serve as the equivalent of adequate physical separation. Areas that come under these considerations include but may not be limited to the following:

Elec. Dwg.#	FHA Dwg.#	Elev.	Area	Div.
202	14	473'5"	Off-Gas	I,II,III
211	13	503'6"	Reactor Bldg (N/W)	I,II
211	13	503'6"	Reactor Bldg (S/W)	I,III
212(+232)	13	496'0"		I,II,III
221	11	525'7"	Reactor Bldg (N/E)	I,II
222	11	525'7"	Reactor Bldg (N/W)	I,II
222	11	525'7"	Reactor Bldg (S/W)	I,II
223	11	536'0"	CSR	I,II,III
223	11	521'0"	Aux. Equip. Rm.	I,II,III
224	11	525'7"	Div. II Switchgear	I,II,III
224	11	525'7"	Lift Truck Route	I,II,III
232	12	510'6"	Div. III Switchgear	I,II,III

234	10	546'0"	Reactor Bldg (S/E)	I,II
235	10	546'0"	Reactor Bldg (S/W)	I,II
237	10	546'0"	Div. 1 Switchgear	I,II
237	10	546'0"	Heater Bay	I,II
240	15	567'5"	HVAC area	I,II
241	15	570'6"	Reactor Bldg	I,II
244		593'6"	Reactor Bldg (S/W)	I,II
264			Various Manholes	I,II,III

At our request, the applicant agreed to perform a fire test to verify the fire rating of the proposed 1-1/2 hour fire barrier design. On June 6, 1979, a fire test was performed by the Portland Cement Association Laboratories (PCAL) Skokie, Illinois. The fire test followed the ASTM E-119 test procedure in a beam furnace and was witnessed by a fire protection consultant to the NRC. The test results are contained in a report prepared by PCAL entitled, "Fire Protective Cable Tray Fire Test" dated June, 1979. The fire test demonstrated conclusively that the proposed fire barrier design is satisfactory and is, therefore, acceptable.

We have reviewed the areas containing redundant divisions of equipment and cable and conclude that with the modifications, the fire protection meets Appendix A to BTP 9.5-1 and is, therefore, acceptable.

F. Diesel Generator Rooms

The diesel generator rooms meet the fire protection guidelines of Appendix A to BTP 9.5-1. However, we were concerned that the location of the exterior air intakes for the Division I and II diesels could be exposed to large amounts of smoke released from an outside transformer fire or from a fire in one of the diesel generator rooms. These air intakes are on the same elevation and located together. At our request, the applicant has agreed to move the Division II air intake to a higher elevation and a different location such that smoke from a fire in the outside transformers or one of the generator rooms would not expose the air intake. That modification rectifies our concern and is, therefore, acceptable.

G. Other Plant Areas

The applicant's Fire Hazards Analysis addresses other plant modifications not specifically discussed in this report. The applicant is installing: additional 8-hour battery pack emergency lighting; portable extinguishers, to include water extinguishers in the control room, cable spreading room, and the electrical equipment rooms; hose stations; portable brigade radios; portable smoke blowers; detectors in control room air intake plenums, class 1E closed cabinets, and on the ceiling of the enclosed area behind the control room main panel; fire/smoke controls for the auxiliary building elevator; sealant for the floor opening around the vent stack; fire rated fire stop materials for replacement of any plastic expansion joints; 3-hour fire barrier and curb to separate the oil tank from the Division III switchgear room; and the fire resistant material on top of cable trays located in control room ceiling. Areas affected by these proposed modifications are in accordance with the guidelines of Appendix A of BTP 9.5-1, and are, therefore, acceptable.

VI. ADMINISTRATIVE CONTROLS AND FIRE BRIGADE

The administrative controls for fire protection consists of the fire protection organization, the fire brigade training, the controls over combustibles and ignition source, the prefire plans and procedures for fighting fires and quality assurance. In revision 6 to the Zimmer Fire Protection Program Reevaluation, the applicant compared the administrative controls to our supplemental guidance "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," dated June 14, 1977. The applicant, at our request, has revised (revisions 1, 6, and 14) their administrative controls and training procedures to meet our supplemental staff guidelines. The applicant will implement the plant administrative controls and procedures before fuel loading.

The applicant had originally proposed a four-man fire brigade. We requested, and the applicant agreed in revision 14, to provide a five-man fire brigade which meets our guidelines, and, therefore, is acceptable.

We conclude that the fire brigade equipment and training conform to the recommendations of the National Fire Protection Association, to Appendix A to BTP ASB 9.5-1, and to our supplemental staff guidelines and, therefore, are acceptable.

VII. CONCLUSION

We find that the Fire Protection Program for the Zimmer Nuclear Plant with the improvements and modifications committed by the applicant to be implemented prior to fuel loading will meet the guidelines contained in Appendix A to BTP ASB 9.5-1. On February 19, 1981, the Commission approved a rule concerning fire protection. Although this rule and its Appendix R are not directly applicable to Wm. H. Zimmer, the requirements set forth in Appendix R are being used as guidelines in the licensing of plants after January 1, 1979. On April 27, 1981, the Commission required that operating licenses issued after January 1, 1979, contain a condition requiring compliance with commitments made by an applicant and agreed to by the staff after differences between the applicants' program and the guidelines set forth in Appendix A to BTP 9.5-1 and Appendix R to 10 CFR 50 have been identified and evaluated.

The applicant has not committed to meet the technical requirements of Appendix R to 10 CFR, Part 50, or provide equivalent protection. Since the review of Wm. H. Zimmer was not specifically conducted to Appendix R requirements, we will require the applicant to meet the technical requirements of Appendix R to 10 CFR, Part 50, or provide equivalent protection.

APPENDIX F

CONTROL ROOM DESIGN REVIEW/AUDIT
SAFETY EVALUATION REPORT
WM. H. ZIMMER

INTRODUCTION

As part of the NRC staff actions following the TMI-2 accident (Item I.D.1, NUREG-0660, Vol. 1, May 1980) (ref. 1), it is required that all licensees and applicants for operating licenses conduct a Detailed Control Room Design Review (DCRDR) to identify and correct human factors design deficiencies. These DCRDRs will be initiated after issuance of NUREG-0700, Guidelines for Control Room Design Reviews (ref. 3), and will be completed within one year. Those applicants for operating licenses who are unable to complete this DCRDR prior to fuel loading are required to conduct a preliminary design assessment of their control rooms, identify human factors deficiencies, and establish a schedule, approved by NRC, for correcting the deficiencies.

As a result of these requirements, Cincinnati Gas and Electric Co. (CG&E) performed a preliminary assessment of the Wm. H. Zimmer control room and submitted its findings to the N . in a report dated February 4, 1981 (ref. 4) for review and evaluation.

A Human Factors Engineering Branch (HFEB) team reviewed the CG&E preliminary assessment report. After reviewing this assessment, the HFEB team, assisted by human factors consultants from Lawrence Livermore National Laboratory and BioTechnology, Inc., conducted an onsite control room audit from February 23 to 27, 1981. All human factors design deficiencies identified and reported by CG&E in their preliminary assessment were reviewed during the HFEB audit to evaluate the suitability of the proposed corrective actions.

Although our review identified some additional human factors deficiencies, we found that the control room was generally designed to promote effective and efficient operator actions. Annunciator panels are used to indicate required operator actions, and will not be used as plant status indicators. Controls are generally well laid out and within easy reach of most operators, and follow generally accepted design conventions for position and direction of movement. Visual displays are generally adequate with respect to design, location and illumination. Controls and displays are generally well organized, and show consideration for functional and sequential arrangement. Control room layout and physical design of the control panels and consoles reflects consideration of human anthropometry. Color has been used effectively, and use of mimics on the panels and consoles enhances operator effectiveness in interfacing with controls and displays.

The NRC Control Room Design Review/Audit report (ref. 5) was forwarded to CG&E April 1, 1981. A meeting was held on April 14, 1981, during which identified deficiencies were discussed, means for the the correction of most deficiencies were resolved, and schedules for correcting deficiencies were established. In subsequent telephone communications with CG&E, all issues were resolved, and a report containing the applicant's commitments (ref. 6) was submitted to NRC.

HUMAN FACTORS DEFICIENCIES IDENTIFIED

The review team identified a number of human factors deficiencies which were documented in a CRDR/Audit report which was transmitted to the applicant. The report categorized the deficiencies according to their importance. Observed human factors design deficiencies were given a priority rating of one, two, or three (high, moderate, low), based on the increased potential for operator error and the possible consequences of that error.

Deficiencies identified as having a high potential for operator error (Category 1) are required to be corrected before loading fuel. Deficiencies given a Category 2 rating must be corrected before 5% power operation. All deficiencies identified as Category 1 or Category 2 are presented in the following sections of this report, along with descriptions of the applicant's commitments to correct these deficiencies.

Deficiencies which were given a Category 3 rating will be addressed by the applicant in the performance of long term studies to determine the best and most feasible solutions. Category 3 items were identified in the CRDR/Audit report, dated April 1, 1981, and are not included in this report.

Throughout the report the use of parentheses, such as (6.5.4), refer to the item number used in the HFEB CRDR/Audit report.

HUMAN FACTORS DEFICIENCIES TO BE CORRECTED BEFORE LOADING FUEL

In the section which follows, the deficiencies given a Category 1 rating in the HFEB CRDR/Audit report are listed with the corresponding CG&E corrective action commitments.

1.0 CONTROL ROOM WORKSPACE

1.1 Annunciators and status displays are difficult to reach for maintenance. Operators cannot change bulbs without standing on the benchboard. (6.1.2)

A platform or ladder device will be provided which will allow safe access to the annunciator panels without interference with the control panels.

1.2 Annunciator bulb replacement has reportedly caused some short circuits which have resulted in fires. (6.1.4)

The short circuits were caused by small "whiskers" protruding from the spot solder connections on the annunciator bulbs. When a deficient bulb was inserted into the socket, the "whisker" was bent or broken, causing a short with potential for fire. Since the manufacturing process cannot be improved, a 100% check of the bulbs will be conducted each time a shipment is received on site.

1.3 Bulb replacement of some Fisher controllers is reported to throw calibration out of tolerance. (6.1.5)

This incident occurred when a 6-volt bulb was placed in the Cooling Tower Makeup Flow Controller. This control, as well as 11 other Fisher KD1101 controls, are used on balance of plant systems. They require a 28-volt bulb. The 6-volt bulb changes circuit resistance which caused a resistor to burn out, resulting in controller failure.

An investigation conducted by plant engineering staff revealed that the 6-volt and 28-volt bulbs are similar in size and shape. However, each bulb is uniquely identified. Further investigation showed that the 6-volt bulb is only used in 12 locations in the control room.

To prevent this situation from developing again, all 6-volt bulbs will be removed from the control room and issued from the store room only under the direct signature of the operating engineer, the assistant superintendent or the station superintendent.

A significant operating event report (SOER) has been developed. The SOER will be reviewed by all licensed operators in the next licensed operator training module.

Since the 6-volt bulbs are only used in 12 locations, a long-term engineering study will be initiated to modify the circuit so that it can accept the 28-volt bulb or modify the bulb socket so that the 6-volt bulb will be a different size or shape.

1.4 There are J-handle switches which are located close to the edge of the benchboards. These may be subject to accidental actuation by operators leaning against the panels. (Example: feedwater controllers on PM03J) (6.1.10)

Guardrails will be installed to prevent accidental activation of Jnandle switches which are near the edge of the benchboard.

1.5 Incoming telephone calls to the plant are transferred to the control room at night. (6.1.13)

During normal operations, routine incoming telephone calls to the plant will not ring in the primary control room area. Routine incoming calls may then be transferred to the primary control room area.

During emergency situations the communicator console located near the center desk area in the control room will be manned by plant personnel, not on duty in the control room, to answer incoming calls to coordinate emergency actions with offsite agencies.

1.6 There are no procedures for communications control during an emergency. (6.1.14)

Procedures directing on-site and off-site communications will be outlined in the site Emergency Plan.

APPENDIX F (Continued)

1.7 Adequacy of communications while using face mask emergency breathing equipment could not be determined. (6.1.19)

An engineering investigation of this problem has been initiated between CG&E and the vendor. The results of the engineering study will determine the method or device required to ensure communications while wearing a Scott Air-Pac.

2.0 WORKPLACE ENVIRONMENT

2.1 The control room HVAC system is not in its final configuration. Climate control should be checked when complete and normal heat sources are operating. Control room air temperature indicators read 70 to 80 degrees (measured at the exhaust ducts). The temperature appeared comfortable, but the operators said that the control room is often too cold. Filtering systems, including those needed to satisfy standard HVAC requirements, are not presently installed and could not be evaluated. (6.2.1, 6.2.2)

Although the control room HVAC system is basically complete from a structural standpoint, final air balancing cannot be accomplished until all electrical penetration seals, fire stops, required filters, etc. have been installed. This incomplete status has resulted in some variance in temperature control. When the final design configuration is complete, the control room HVAC system will be rebalanced and reevaluated to assure the system performs in accordance with established design criteria. This reevaluation will include noise testing, temperature control, and filter system requirements. The HVAC system pre-operational test will satisfy this requirement.

2.2 Ambient illumination is well below recommended levels. (6.2.3)

The normal lighting level will be increased to a minimum of 30 foot candles for direct lighting.

2.3 Illumination levels vary excessively throughout the control room. (6.2.4)

The normal lighting level in the control room will be increased to a minimum of 30 foot candles for direct lighting and balanced appropriately.

2.4 Illumination levels are not controllable in areas of the control room where close work must be done. (6.2.5)

Desk lamps will be provided for those areas of the control room where close work must be done.

2.5 The emergency backup lighting system does not provide the recommended illumination levels. (6.2.7)

The emergency AC lighting level will be increased to meet the 30 foot candle minimum criteria for direct lighting. The DC emergency lighting will be increased to meet the 3 foot candle minimum criteria.

2.6 Emergency lighting is not provided in the immediate area of the remote shutdown panels. (6.2.8)

Emergency AC and DC lights are provided near the remote shutdown panels. In addition, the remote shutdown panels are equipped with fluorescent lights just inside the doors which provide sufficient light to operate the panels. These fluorescent lights will be added to the emergency lighting system.

2.7 There is excessive glare on the labels throughout the control room. (6.2.11)

All labels in the control room will be replaced using non-glare material. Black lettering on white background will be used.

2.8 The "C" annunciator audio signal (measured at the horn) was 8 db below the average ambient noise level of the surrounding area and was inaudible to control room personnel. (6.2.15)

After appropriate modifications have been made to the annunciator system (i.e. directional audibility) and the control room HVAC system has been finally air balanced and noise criteria satisfied, the auditory alarms will be set at a minimum of 10 db above the average ambient noise level.

2.9 The computer system audible alarm has a variable level control which was turned down during the audit. When the intensity of this alarm is adjusted down, it is detectable only in the immediate vicinity of the computer console as a 60 cycle/second hum. (6.2.17)

After appropriate modifications have been made to the annunciator system (i.e. directional audibility) and the control room HVAC system has been finally air balanced and noise criteria satisfied, the computer system audible alarm will be replaced so that the minimum setting is at least 10 db above the average ambient noise level.

2.10 There is no lamp test capability and no use made of dual filament bulbs for failure indication (except for the annunciators). (6.2.21)

An engineering investigation of this problem has been initiated. Long-life light bulbs will be installed and burn-in criteria will be established. An operational review, performed daily, to document the status of bulbs that are supposed to be illuminated will be incorporated before fuel load. Verification of other bulbs will be made during surveillance operations. A long term study will be initiated to find a solution to this problem.

2.11 Tag-outs are not securely attached to some of the vertical push-buttons. (6.2.24)

The tags have pre-cut holes to provide an easy and secure method of attachment. The tagging directives will be revised to instruct the operators to loosen the collar of the pushbutton, insert the tag, and retighten the collar to ensure that the tag stays in place.

3.0 ANNUNCIATORS AND ALARMS

3.1 Annunciator window covers are not keyed or matrix coded to prevent accidental interchange when they are removed for bulb replacement. (6.3.3)

The annunciator window frame, where the window tile is an integral part of the frame, will be matrix coded to prevent accidental interchange.

3.2 There are no formal procedural requirements for testing and inspection of visual annunciators. (6.3.6)

The testing and inspection of visual annunciators will be completed on a daily basis. This requirement will be formally promulgated in the Station Operations Directive, OS.SAD.01, and documented on the Shift Turnover Checklist.

3.3 There is presently no system of prioritization of visual annunciators according to severity. (6.3.8)

The following visual coding scheme is to be followed for the prioritization of annunciators:

COLOR CODE	ANNUNCIATOR CLASSIFICATION
Red	Used to denote emergency conditions which require operator action to be taken without undue delay to avert impending personnel injury, equipment damage, or both.
Amber	Used to advise an operator that a condition exists which is marginal, or to alert the operator to situations where caution, recheck, or unexpected delay is necessary.
Blue	Used to indicate system conditions that do not have "right" or "wrong" implications, such as alternative functions (e.g., Pump #1A selected for use, etc.) or transitory conditions (e.g., action or test in progress, function available), provided such indication does not imply success or failure of operations.
Green	Used to indicate that the monitored equipment has returned to satisfactory tolerance envelopes.

3.4 On many annunciator windows, the characters are too small (less than 3/8") for easy reading at the required distances. (6.3.10)

New annunciator windows will be engraved for the entire control room to ensure the readability of all primary messages from the location of the associated annunciator response controls.

3.5 Different print styles are sometimes used on adjacent annunciator windows. (6.3.11)

New annunciator windows will be engraved for the entire control room by a single manufacturer to ensure that a universal print style is used.

3.6 On many annunciator windows, messages are crowded with inadequate spacing between characters and between lines, making reading difficult. (6.3.12)

New annunciator windows will be engraved for the entire control room that will provide adequate spacing.

3.7 Annunciator messages are frequently too long and wordy, contributing to message crowding. Some messages are non-specific and ambiguous. (6.3.13)

The text of all annunciator windows will be reviewed for correctness. New windows will be engraved using the Standardized Abbreviation List developed for the control room.

3.8 Many annunciator windows have temporary titles printed on tape. These are especially difficult to read when the window is illuminated. (6.3.14)

New windows will be engraved for the entire control room using the Standardized Abbreviation List.

3.9 Many abbreviations are used, but abbreviations are not consistent from one window to the next. Some contain misspelled abbreviations. (6.3.15)

The text of all annunciator windows will be reviewed for correctness. New windows will be engraved using the Standardized Abbreviation List.

3.10 There is no distinctive audio evacuation signal available. Presently, an evacuation announcement is made over the P.A. system by the Shift Supervisor (6.3.20)

The Thompson P.A. system is equipped with six different audible warning tones. One has been designated as the "Site Evacuation" alarm. These alarms are to be tested during the pre-operational test of the system, after construction is complete.

4.0 CONTROLS

4.1 A "moving scale - fixed index" rotary control knob was found in the control room that did not have an index pointer. (6.4.18)

A pointer will be added to indicate the selected position of this rotary control knob.

5.0 VISUAL DISPLAYS

5.1 The Process Radiation Monitor inverter had one missing indicator light and one burned out light. (6.5.6)

The Process Radiation Monitor inverter "power on" lights will be operational when the pre-operational test is complete.

6.0 PANEL LAYOUT

No deficiencies required to be corrected before fuel load.

7.0 CONTROL-DISPLAY INTEGRATION

7.1 Controls and displays for the containment isolation system are physically separated. They presently exist on three panels: P632, P601, and PM07J. (6.7.5)

The drywell equipment drain and floor drain valve indications, controls, and sump pump flow indicators and the Primary Containment Isolation reset buttons with the associated indicator lights will be moved to panel P601. In addition, a containment mimic will be installed on panel P601. The mimic will include indications of containment group isolations. Containment parameter indicators will be located near the mimic. Human engineering considerations will be used to determine these locations. The hardware and board modifications will be completed by fuel load. The mimic will be operational by full power.

8.0 LABELS AND LOCATION AIDS

8.1 Temparary tape mimic labels do not adhere well to the benchboard. (6.8.9)

Non-permanent labels will be replaced. However, it is recognized that the use of non-permanent labels is necessary during plant start-up because they supply useful information to the licensed operators during plant construction and testing. An administrative procedure controlling and recording the use of non-permanent labels and their replacement with permanent labels will be established.

8.2 The label on the Steam Line Inboard Drain switch on P602 is handwritten. (6.8.11)

A new label will be installed using the Standardized Abbreviation List.

8.3 The mimic flow indication below the Reactor Water Cleanup System Conductivity recorders is in conflict with the recorder labels. The outlet recorder is located to the left of the inlet recorder. The mimic shows the inlet to the left of the outlet. (6.8.13)

The recorder positions will be swapped.

8.4 The RHR Shutdown Cooling Injection Valve (E12-F053B) has a yellow label (meaning Division 1) but has a tape label above (in blue) which reads Division 2. (6.8.25)

The RHR Shutdown Cooling Injection Valve (1E12-F053B) connects RHR loop 1B to Reactor Recirculation loop 1A. This valve is powered from RXMCC 1A, which is a Division 1 bus The blue tape label will be removed.

8.5 Mimics do not have primary and secondary paths indicated. (6.8.26)

A consistent scheme of color and shape coding will be used on all mimics. Mimics will be made of colored tape that adheres well to the panel surface. Thick lines (1/4") will indicate primary flow paths and thin lines (1/8") will indicate secondary paths.

9.0 PROCESS COMPUTER

No deficiencies required to be corrected before fuel load.

10.0 DATA RECORDING AND RETRIEVAL

10.1 Calibration stickers obstruct the view through some recorder faces. (6.10.3)

Extraneous material will be removed from recorder faces. Calibration stickers will be relocated so that the view through the recorder faces will be unobstructed.

10.2 A strip chart recorder was found that had paper with a different printed scale than the scale on the recorder. (6.10.6)

Recorder labels will be marked to indicate correlation with pen color, and to ensure that chart paper with the correct scale is used.

DEFICIENCIES TO BE CORRECTED AFTER LOADING FUEL AND BEFORE 5% POWER

In the section which follows, the deficiencies given a Category 2 rating in the HFEB CRDR/Audit report are listed with the corresponding CG&E corrective action commitments.

1.0 CONTROL ROOM WORKSPACE

1.1 Telephones can be easily knocked off their cradles by operators standing at the panels. (6.1.12)

The phones or their holders will be modified, relocated or replaced to ensure that they are not easily knocked off their cradles.

1.2 There are no paging system loud-speakers at the remote shutdown panels. (6.1.18)

A Thompson P.A. speaker and handset will be installed at or adjacent to the Remote Shutdown Panels 1PL67JA and 1PL67JB.

1.3 The procedures books are not organized so that the operators can easily identify the location of all needed procedures. (6.1.20)

Complete sets of operating procedures will be located at each of three stations in the main control room. Operators will be trained as to their access, dedication, and use.

2.0 WORKPLACE ENVIRONMENT

2.1 The red status panel on PM03J has a luminance ratio (LR) of 25:1 to the adjacent area (20:1 recommended max). (6.2.12)

The WILB on PM03J will be modified to meet the maximum luminance ratio criteria of 20:1.

2.2 The scram group system A & B white indicator lights have a LR of 37:1 to the adjacent area (20:1 recommended max). (6.2.13)

The scram indicator lights will be color coded with a blue lens which will reduce the LR. The LR will be checked after this change to determine whether a further, long term solution is necessary.

2.3 The white status panel on PM02J has a LR of 420:1 to the adjacent area (20:1 recommended max). (6.2.14)

The Feedwater Heater Drain Valve status panel on PM02J will be modified to meet the maximum luminance ratio criteria of 20:1.

2.4 The annunciator "Reset Chime" was measured to be only approximately 3 db above the average ambient noise level. (6.2.16)

After appropriate modifications have been made to the annunciator system (i.e. directional audibility) and the control room HVAC system has been finally air balanced and noise criteria satisfied, the annunciator "Reset Chime" will be set a a minimum of 10 db above the average ambient noise level.

2.5 The paging system was only approximately 5 db above the average ambient noise level. It was extremely difficult to discriminate messages over the paging system. This was confirmed during the procedures walkthrough. (6.2.18, 6.2.19)

The Thompson P.A. system in the control room will be adjusted to ensure full speech intelligibility at all points within the main control panel circle.

2.6 Tag-outs obscure labels, legends, indicator lights, and annunciator windows. (6.2.23)

The annunciator tag-out forms will not be used after the start of operations. The other tag-out forms will be reduced in size so that they no longer can obscure associated annunciator lights.

3.0 ANNUNCIATORS AND ALARMS

3.1 While most visual annunciators are located above their related controls and displays, there are some exceptions. (6.3.1)

MSIV Leakage Control annunciators will be moved to control panels P654 and P655. In addition, containment parameter annunciators will be moved from back panel PM06J to panel P601.

3.2 All annunciator windows presently:

- flash red-white to announce,
- are steady red after acknowledgment, and
- are steady green when cleared.

Red color for all annunciator tile windows is poor usage. There also are significant differences in shading from tile to tile. (6.3.17)

The annunciator visual coding scheme will be changed according to the color code described in item 3.3 of the previous section. The significant differences in shading is due to the use of different manufacturers and material. The annunciators will be re-engraved by a single manufacturer on similar material for the entire control room.

3.3 No annunciator panel localization is provided by the auditory alert horns or clear signal chimes. The operator must locate alarm indications by scanning visual annunciators. (6.3.18)

An auditory coding scheme consisting of four separate sound sources will be installed. Each sound source will be located in its respective control panel near the annunciator windows. This coding scheme will enable the operator to quickly identify the sound source (e.g., control panel subsystem) without having to scan all the annunciator tiles.

3.4 Auditory signals are not differentiable between critical and non-critical alarms, or between types of noncritical alarms such as warnings, cautions, or alerts. (6.3.19)

The directional auditory signals described in item 3.3 of this section direct the operator to a local area of the control room. The operator can then differentiate between the critical and non-critical alarms by the prioritization color code for the annunciators as described in item 3.3 of the previous section.

4.0 CONTROLS

4.1 Some bar switches on the remote shutdown panels have to be held actuated for up to two minutes. This is reported to be very difficult and fatiguing. (6.4.1)

The control switch coding adopted by the applicant provides an adaptor known as a "glove" which is permanently mounted on the throttle valve bar switches. The "glove" provides sufficient leverage so that fatigue felt during actuation of these switches is significantly reduced.

4.2 Bar switches cannot be readily operated while wearing gloves. These switches are used on the remote shutdown panels, as well as several other locations where personnel may be required to wear emergency protective clothing, including gloves. (6.4.2)

Bar switches used to operate throttle valves will be provided with a "glove" adaptor. This adaptor is a permanently installed device and provides the operator with sufficient surface area so that operation with protective clothing gloves may be accomplished without difficulty.

4.3 Rod control push buttons require excessive pressure to operate. This produces fatigue in the operator's hand after several minutes of operation. This problem is amplified when the operator is seated. (6.4.3)

A mushroom handle will be developed so that the operators may apply pressure with the hand instead of a finger. This handle will not be integral with the pushbutton.

4.4 Pushbuttons on the Bailey and General Electric controllers are too small. Some of these pushbuttons also are flush with the controller surface. (6.4.4)

The Feedwater Level Controller Auto/Manual pushbuttons on panel 603 will be raised by extending the button surface. All other Bailey and General Electric controllers will be checked to ensure that they are easy to operate.

4.5 The pushbuttons on the remote shutdown panel do not provide any activation feedback such as an audible click or the activation of an indicator light. (6.4.5)

Initiation of the Automatic Depressurization System (ADS), subjects the reactor to rapid depressurization. The operator is given positive feedback by observing the rapid decrease in reactor pressure, swell in reactor water level, and increase in suppression pool level and water temperature.

4.6 Both of two transfer switches must be activated to transfer indication to the remote shutdown panels. This should be indicated on the panels. (6.4.9)

Labels will be installed to indicate that both switches must be activated to transfer indication to the remote shutdown panels.

4.7 The movements for on/off actuation of some controls on panels PM07J and PM08J violate plant convention. (6.4.16)

The Drywell Pneumatic Compressor Control Switches have positions as follows:

BASE/STANDBY/OFF

The control switches will be rewired to follow the following convention:

OFF/STANDBY/BASE

4.8 The "purge" position on the post-LOCA selector on PM07J is the "both" position but is not indicated. (6.4.20)

The control switch escutcheion will be revised to provide the following terminology:

"Relief"- Only 1VQ005A & B valves are open to relieve drywell pressure.
"Closed"- The 1VQ005A & B valves are closed.
"Purge" - Only 1VQ005A & B valves are open to relieve drywell pressure.

5.0 VISUAL DISPLAYS

5.1 Dirt and damage on the plastic cover of a remote shutdown panel meter obscures the scale. (6.5.1)

The meter faces will be cleaned of extraneous material, repaired, or replaced.

5.2 Some display information required during normal operation is currently located on back panels. (6.5.4)

The Drywell equipment drain and floor drain valve indications, controls, and sump pump flow indicators and the Primary Containment Isolation reset buttons with the associated indicator lights will be moved to panel P601. The planned containment mimic will include indications of containment group isolations. Containment parameter indicators will be located near the mimic. Human engineering considerations will be used to determine these locations.

5.3 Normal operating limits are not generally marked with permanent markings on indicator scales. (6.5.7)

The normal operating limits are found in Operating Log Sheets and System Operating Procedures. However, to increase operator response capability, green transparent tape will be added to the external surface of selected meters which have a normal operating range. A red or amber horizontal line of tape may be added to denote a trip or alarm setpoint, respectively. If the tape is helpful to the operators, it will be applied permanently under the meter pointers prior to beginning the second fuel cycle.

5.4 A white indicator light is used for a trip condition on the remote shutdown panel. Plant convention is yellow/amber for trip and white for a general alarm. (6.5.10)

The RCIC Turbine Trip light on the Remote Shutdown Panel will have the white lens replaced by an amber lens.

5.5 Remote shutdown panels lack any mimics, demarcation lines, or functional group labeling to enhance recognition and identification. (6.5.11)

Relabeling, knob coding, lines of demarcation and color padding will be used to improve these panels.

5.6 Groups of meters in horizontal strings of 5 or more on PM07J are not broken up into smaller groups. (6.5.12)

Visual demarcation will be provided to the meters through the use of color pads, lines, and/or global labeling to highlight the meters pertaining to each particular subsystem.

5.7 Bailey controllers and some Fisher controllers have scales that do not indicate units. (Example: Dump Valve Manual Control Indicator) (6.5.15)

Units will be added to the following controller scales: Dump Valve Manual controller, Control Rod Drive System controller, and Feedwater controller.

5.8 Some meters give information that must be converted to different units before use. (6.5.16)

The Reactor Recirculation loop flow indicators 1B33-R611A and B measure flow in gpm. The flow meter will be changed to read percent flow.

6.0 PANEL LAYOUT

No deficiencies required to be corrected before going to 5 percent power.

7.0 CONTROL-DISPLAY INTEGRATION

7.1 Demarcation of associated controls and displays was not used sufficiently. (6.7.10)

Demarcation and/or color padding for control room panels are shown in Figures 1 through 6, Appendix D of the applicant's "Preliminary Assessment Human Factors Review of the Wm. H. Zimmer Nuclear Power Station Control Room" report (ref. 4). In addition, the relabeling, knob coding, lines of demarcation and color padding of panels PM07J, PL67JA, and PL67JB will be used to improve these panels.

8.0 LABELS AND LOCATION AIDS

8.1 No panel designation labeling is provided. (6.8.1)

Panel labels will be installed using the Standardized Abbreviation List.

8.2 The location of labels on some panels is not consistent (some above, some below). (6.8.2)

The following convention will be followed for control/display labeling:

a. All Panels - Black lettering on a white background.

b. Vertical Panels - Labels will be placed above controls and displays. Global labels for subsystem, module, or component identification will be located above all controls and displays.

c. Benchboard Panels - Labels will be placed above controls and displays. Global labels for subsystem, module, or component identification will be above all controls and displays.

8.3 There are no labels on the outside of either remote shutdown panel. (6.8.3)

Panel labels will be installed using the Standardized Abbreviation List.

8.4 There are no warning labels on the front of the remote shutdown panel doors to warn that the doors are alarmed. (6.8.4)

Panel warning labels will be installed using the Standardized Abbreviation List.

8.5 System labels for individual systems are located between displays and controls on P601 and P602. The labels should be above the displays. (6.8.7)

System labels will be placed above the indicators on the vertical section of P601 and P602.

8.6 Labels for the alarm lights on the remote shutdown panel should be located closer to the lights than they are presently. (6.8.8)

The labels will be located closer.

8.7 The lettering has rubbed off of the Bailey Feedwater level controller because the painted letters were not covered by plastic. (6.8.10)

Lettering will be refilled with white paint and then covered with a clear plastisol coating to prevent lettering from being rubbed off again.

8.9 Label content on the remote shutdown panel is not clear and uses inconsistent terminology. (6.8.12)

New labels will be installed using the Standardized Abbreviation List.

8.10 Label abbreviations are not standardized. Some label abbreviations are used for different meanings at different locations. (6.8.14)

New labels will be installed using the Standardized Abbreviation List.

8.11 Some controls are identified by both G.E. and S&L numbers. (6.8.16)

New labels will be installed using the Standardized Abbreviation List. The S&L numbers will be used.

8.12 A label was found that read "Turning Gear Engaged". It should have read "Turbine Zero Speed". (6.8.17)

A new label will be installed with the correct legend.

8.13 Wording on labels is not always brief and concise. (6.8.18)

Labels will be reviewed to ensure that the wording is brief and concise. New labels will be installed using the revised wording.

8.14 Some labels do not have at least one stroke width between characters. (6.8.19)

The following convention is to be followed for stroke width:

BIT SIZE	LETTER SIZE
.015"	(Small 1/8", 5/32")
.030"	(Large 1/2", 3/8", 1/4", 3/16")

(Engraving Bit Sizes .015" and .030" are approximate.)

8.15 Color coding of labels is used inconsistently where any attempt is made to use it at all. (6.8.20)

New labels will use black letters on white background. A color coded dot will be used to denote electrical divisions.

8.16 All Division 1 labels should be yellow to be consistent with the present control room color scheme. (6.8.23)

All labels in the control room are to be replaced with black lettering on white background. A color coded dot will be used to denote electrical divisions.

8.17 Not all mimics indicate direction of flow and those that do are often confusing. (Examples: Off Gas system mimic, Feedwater System mimic) (6.8.27)

Direction of flow will be indicated on mimics.

8.18 Color coding of mimics is inconsistent and incomplete. (6.8.28)

A consistent scheme of color and shape coding will be used on all mimics.

9.0 PROCESS COMPUTER

9.1 All process parameter addresses must be referenced in an index list kept near the console. These points are not cross-indexed by name, system/subsystem, or functional group. (6.9.2)

A cross-index will be prepared which lists process parameters by system/subsystem, by name, and by point I.D. for ease of reference. The cross-index will be maintained at the computer operator's console.

9.2 The process computer printouts are difficult to read. The printing is too light and lines are too closely spaced. (6.9.8)

The printer has a toggle switch which permits printouts to be single or double spaced. During transient or accident conditions, the output typers can be operated in the double space mode as deemed necessary by the operator.

Light printing is caused by worn typing ribbons. The preventive maintenance schedule for the process computer will be changed to require more frequent changing of output typewriter ribbons.

9.3 The alarm acknowledge and the action buttons are next to each other and have no coding to distinguish between them. (6.9.11)

The alarm acknowledge pushbutton will be retrofitted with a colored collar, consistent with the color coding scheme, to distinguish it from the action pushbutton.

9.4 Abbreviations used by the process computer are not always consistent with the master list. (6.9.12)

A cross-reference list will be developed to match the Standardized Abbreviation List with the appropriate abbreviation on the function map of the operator's console. Standardizing the abbreviations used by the computer would entail considerable software revision which is not considered to be warranted at this time.

A new process computer will be installed at the first refueling outage. The abbreviations used in the software for the new computer will follow the Standardized Abbreviation List.

10.0 DATA RECORDING AND RETRIEVAL

10.1 Recorders on the Radiation Monitor panels have no scale parameter names. (6.10.2)

Scale parameter names will be added to these recorders.

10.2 Many recorder charts do not have time references. They do not give sufficient information to ensure correct interpretation. (6.10.4)

Chart paper with time reference information will be installed.

CONCLUSIONS

Items identified in our CRDR report as category 3 items have not been addressed in this report. We require the applicant to address these items as well as other deficiencies that may be identified in their detailed CRDR (NUREG-0700), and will expect final resolution of all deficiencies on a schedule consistant with NUREG-0737.

Based on our review of the licensee's submittals, our control room review, and other clarifying information, we conclude that with the corrections required before startup and before 5 percent power, the potential for operator error leading to serious consequences as a result of human factors considerations in the control room is sufficiently low to permit startup and power operations of the William H. Zimmer Nuclear Plant.

APPENDIX F (Continued)

REFERENCES

1. NUREG-0660, Volume 1, May 1980; NRC Action Plan Developed as a Result of the TMI-2 Accident.*

2. NUREG-0737, November 1980; Clarification of TMI Action Plan Requirements.*

3. NUREG-0700, Guidelines for Control Room Design Review. (in preparation, to be published July 1981)

4. Preliminary Assessment Human Factors Review of the Wm. H. Zimmer Nuclear Power Station Control Room, February 4, 1981, The Cincinnati Gas & Electric Co.

5. Letter: Voss. A. Moore to A. Schwencer, April 1, 1981; Control Room Design Review/Audit Report - W. H. Zimmer.

6. Letter: E. A. Borgmann to Harold R. Denton, May 1, 1981; Response to the NRC's Human Factors Engineering Control Room Design Review/Audit Report, Wm. H. Zimmer Nuclear Power Station.

*Available free upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

APPENDIX G

EVALUATION OF LICENSEE EMERGENCY PLAN

EMERGENCY PREPAREDNESS EVALUATION REPORT

BY THE

DIVISION OF EMERGENCY PREPAREDNESS

OFFICE OF INSPECTION AND ENFORCEMENT
ZIMMER NUCLEAR POWER STATION

DOCKET NO. 50-358

MAY 1981

INTRODUCTION

The Nuclear Regulatory Commission's (NRC) evaluation of the state of emergency preparedness associated with the Zimmer Nuclear Power Station involves review of the licensee's onsite emergency plans plus review of the Federal Emergency Management Agency (FEMA) findings and determinations pertaining to State and local emergency preparedness. This evaluation report addresses the licensee's emergency preparedness. A subsequent supplement to this report will address the FEMA findings and determinations providing an evaluation of the status of emergency preparedness associated with the Zimmer site.

The Cincinnati Gas and Electric Company, Columbus and Southern Ohio Electric Company, and the Dayton Power and Light Company (hereinafter referred to jointly as the Licensee) filed with the NRC a comprehensive revision to the Zimmer Nuclear Power Station Emergency Plan (Plan) in January 1981. Previously, the staff had reviewed preliminary versions of the Plan, conducted a site visit to the facility, and held a local public meeting on emergency preparedness.

The January 1981 Plan was reviewed against the 16 planning standards in 10 CFR 50.47, the requirements of 10 CFR 50, Appendix E, and the specific criteria of NUREG-0654/FEMA-Rev. 1 entitled, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, November 1980.

This evaluation report follows the format of Part II of NUREG-0654. Each of the Planning Standards is listed and is followed by a summary of applicable portions of the Plan that relate principally to that specific standard. The conclusions of the staff review are provided at the end of this report.

EVALUATION OF LICENSEE EMERGENCY PLAN

EVALUATION

A. Assignment of Responsibility (Organization Control)

Standard

Primary responsibilities for emergency response by the nuclear facility licensee and by State and local organizations within the Emergency Planning Zones (EPZ) have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

Licensee Emergency Plan Evaluation

The Shift Supervisor for each unit of the Zimmer Nuclear Plant is initially designated as the Emergency Director. When an abnormal condition arises, it is his responsibility to determine if the abnormality meets any of the emergency classifications specified in the plan and to implement the plan, if necessary. There is a 24-hour-a-day communication capability between the station and Federal, State, and local response organizations to ensure rapid transmittal of accurate notification information and emergency assessment data.

Responsibility for overall performance of the emergency response organization is vested in the Emergency Director who is responsible for the overall direction of the plant emergency organization. Qualified members of the station staff who report directly to the Emergency Director have been assigned specific responsibilities for the major elements of emergency response.

Resolution of the Following is Needed

Updated written agreements with appropriate agencies and organizations need to be maintained. These agreements should be rewritten to provide concepts of operation, specific support commitments, authorities' responsibilities and limits on actions of contractors, private organizations, and local services support groups.

B. Onsite Emergency Organization

Standard

On-shift facility licensee responsibilities for emergency responses are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.

Licensee Emergency Plan Evaluation

The Duty Shift Supervisor (or Senior Control Operator) assumes the Emergency Duty Supervisor (EDS) function until the appointed EDS arrives. Qualified station personnel who assume the EDS role are on call on a weekly basis and available within 30 minutes. These individuals assume the EDS role until the Station Superintendent arrives. The authorities and responsibilities of the Emergency Duty Supervisor are clearly defined, specifying those responsibilities that cannot be delegated. The EDS can immediately and unilaterally declare an emergency and make offsite notification.

Station staff emergency assignments have been made and the relationship between the emergency organization and normal staff complement are specified in the plan. Positions and/or titles of shift and plant staff personnel, both onsite and offsite, assigned emergency functional duties are listed.

Resolution of the Following is Needed

The plan does not indicate that minimum staffing requirements, as per Table B-1 of the criteria, will be established. Specifically, only seven qualified individuals are available on a 24-hour/day basis, as per Table 5-2. A total augmentation plan should be established showing how staffing requirements and the 30- and 60-minute augmentation schedule will be met. The total plan augmentation specifies only 14 people; Table B-1 lists 28. The augmentation plans should also address measures to be taken to expedite augmentation during inclement weather.

C. Emergency Response Support and Resources

Standard

Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and local staff at the licensee's near-site Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

Licensee Emergency Plan Evaluation

Arrangements for requesting and utilizing outside resources have been made, including authority to request Federal assistance, as well as assistance from the reactor vendor and the architect/engineer by corporate headquarters support personnel. The Emergency Operations Facility (EOF) will be activiated for the more serious emergency classifications having or potentially having environmental consequences (Alert, Site Area Emergency, and General Emergency). The EOF will accommodate representatives from Federal, State, and local government agencies, as well as representatives from contractor and other support groups. It will be the central data collection point for providing information needed by primary response agencies for implementation of offsite protective actions.

Resolution of the Following is Needed

1. Provide for incorporating the Federal response capability i.e., Department of Energy. The Plan should specify the Federal resources expected for accident categories in Appendix 1 of the criteria, including expected times of arrival at the site. Specific licensee resources as needed to support the Federal response should be listed in the Plan, e.g., airfields, command posts, telephone lines, radio frequencies, and telecommunications centers.

2. Identify available radiological laboratories and their capabilities, and the expected response times of support groups that can be used in an emergency i.e., vendors, universities, private laboratories, etc.

D. Emergency Classification System

Standard

A standard emergency classification and action level scheme, the basis of which include facility system and effluent parameters is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial response measures.

Licensee Emergency Plan Evaluation

The four standard emergency classes (Unusual Event, Alert, Site Area Emergency, and General Emergency) have been established by the licensee. Emergency Action Levels (EALs) are established based upon onsite and offsite radiation monitoring information and upon readings from various reactor sensors. These EALs are used for rapid classification of emergency situations. The EALs are observable and measurable, and are identified using specific instrumentation, parameters, and equipment status. The emergency classification and action level scheme is consistent with the criteria of Appendix 1 to NUREG-0654.

E. Notification Methods and Procedures

Standard

Procedures have been established for notification, by the licensee of State and local response organizations and for notification of emergency personnel by all response organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instructions to the populace within the plume exposure pathway Emergency Planning Zone have been established.

Licensee Emergency Plan Evaluation

Procedures have been established for notification of State and local response organizations in case of emergency. The Emergency Duty Supervisor has been given the authority and responsibility to make prompt notification to declare an emergency.

Resolution of the Following is Needed

1. A Prompt Alerting and Notification System meeting the design objectives of Appendix 3 of the criteria must be developed and installed. The Plan should address the administrative and physical means, and the time required to promptly notify the public of an emergency.

2. Provide for written messages intended for the public, consistent with the operator's classification scheme. In particular, messages to the public giving instructions regarding specific protective actions to be taken by occupants of affected areas should be included in the licensee's plan. In addition, the plan should contain a description of the message authentication scheme and verification procedures.

F. Emergency Communications

Standard

Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

Licensee Emergency Plan Evaluation

The communication system is designed to provide redundant and diverse communications to essential onsite and offsite locations. Within the plant, there are a multi-channel, hard-wired paging system, a PSBX telephone system, and a two-way radio system. Communications between the station, State, and county EOCs are provided by telephone lines with a backup private microwave. Communications between the station and the Clermont County Sheriff will be via microwave while communications with other police agencies, hospitals, water departments, U.S. Coast Guard, aircraft, and the New Richmond School will be via two-way radio. The microwave system will provide the main interface with the Cincinnati central Bell Telephone Office.

Resolution of the Following is Needed

1. A description stating how State, local, and other support groups will be notified 24 hours per day (e.g., town sheriff, volunteer fire fighters, local EOC volunteers).

2. A description of the communications link between the facility and mobile medical support.

G. Public Information

Standard

Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency; the principal points of contact with the news media for dissemination of information during an emergency (including physical location or locations) are established in advance; and procedures for coordinated dissemination of information to the public are established.

Licensee Emergency Plan Evaluation

The utility is in the process of developing a comprehensive plan for public information dissemination.

Resolution of the Following is Needed

1. The Public Information Program indicated in the plan does not clearly define who in the public will receive periodic information regarding how they will be notified, what their actions should be in an emergency, the agreed-upon means of evacuation verification, the location of relocation centers, and the use of radioprotective drugs. Further, the program does not make provisions for the special needs of the handicapped.

2. Include an actual sample of the Public Information Program that will be distributed. This Program will be reviewed by the NRC and FEMA to determine that it meets the planning objective.

3. Indicate that space will be made available for the news media at the nearsite EOF, and describe the annual training program for news media.

4. Define the methods to be used for rumor control.

5. Specify where the information offices will be located.

H. Emergency Facilities and Equipment

Standard

Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

APPENDIX G (Continued)

Licensee Emergency Plan Evaluation

Emergency facilities needed to support an emergency response have been provided including an interim Technical Support Center (TSC), an Operations Support Center (OSC), and an interim Emergency Operations Facility (EOF). Each will be activiated for an Alert or higher emergency classification.

The permanent TSC is currently under construction on the ground floor of the new addition to the Service Building. The temporary TSC is located on the second floor of the service building in a conference room. Both the temporary and permanent TSC's are within a 2.5-minute walking distance from the Control Room. The permanent TSC is designed to have the same radiological habitability as the Control Room under accident conditions.

The Operations Support Center consists of three centers; one located adjacent to the Control Room of the Auxillary Building; the second located at the east end of the Service Building on the main floor, the third in the Maintenance Shop, in the approximate center of the Service Building.

The interim Emergency Operations Facility (EOF) is located in the Moscow Elementary School which the utility has leased and is approximately one-half mile from the Control Room. The EOF is where continued evaluation and coordination of licensee activities related to the emergency will be carried out.

Onsite monitoring systems and instrumentation used to initiate emergency measures and/or provide continuing assessment are identified. They are a meteorology system with wind speed and direction and temperatures capability; seismic instrumentation to measure ground acceleration levels; installed process radiation monitors to measure upward deviations in radiation levels in process lines that actually or potentially contain radioactive effluents; installed area radiation monitors to measure upward deviations in radiation levels in specific locations in the station; fire and smoke detection instruments placed in strategic plant locations; portable dose rate and radiation detection instruments and laboratory counting and analysis facilities.

Resolution of the Following is Needed

1. Provide for "as built" diagrams for use by personnel in the EOF, TSC, and OSC.

2. Specify the types of equipment available in the TSC and EOF, including the types and locations of communications equipment. This information should be detailed on a scaled drawing.

3. Provide the road distance and travel time between the control room and nearsite EOF.

4. Identify what percentage of the personnel will have protective equipment available to them during an emergency.

5. Describe the meteorological instrumentation and procedures which satisfy the criteria in Appendix 2, and the provisions to obtain representative real-time meteorological information from other sources.

6. Clearly identify the provisions for inspection, inventory, quarterly operational checks and calibration of both fixed and portable instruments and equipment (protective equipment, communication equipment, radiological monitoring equipment, and emergency supplies).

7. Identify laboratory facilities, their capabilities, and expected backup response that could be used during an emergency.

I. Accident Assessment

Standard

Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

Licensee Emergency Plan Evaluation

The plan describes systems, equipment, and facilities that will be used as natural phenomena monitors, process radiation, containment monitoring systems (High range and Post-Accident), secondary containment radiation monitors, stack monitoring systems, etc.

The applicant states that there is onsite capability and resources to provide initial and continuing assessment throughout the course of an accident, read-out in the control room, post-accident sampling capability, and containment monitoring.

Onsite and offsite surveys will be performed to verify release information or will be used as a backup assessment method, should the instrumentation used for dose assessment go offscale or become inoperable. An environs survey team can be placed in the field within one hour.

Resolution of the Following is Needed

1. Include a plot or graph indicating the relationship between the containment radiation monitor(s) reading(s) and the radioactive material available for release from containment.

2. Establish methods and techniques to determine the magnitude of a release of radioactive materials based on plant efffluent monitors. In addition, establish the relationship between effluent monitor readings and onsite and offsite exposures and contamination for various meteorological conditions.

3. Describe the methodology for determining release rates and projected doses if the instrumentation used for assessment were to go offscale or become inoperable.

4. Describe the capability and resources for field monitoring within the plume Emergency Planning Zone including the methods, equipment, and expertise to make rapid assessments of the actual or potential magnitude and location of any radiological hazards through the liquid or gaseous pathways. The

description should address activation criteria, means of notification, field team composition, transportation, communication, and monitoring equipment.

5. Describe the means for relating measured field contamination levels to dose rates for key isotopes and gross radioactivity measurements. The plan should also describe provisions for estimating an integrated dose from these estimates with protective action guides.

6. Describe the provisions for individual respiratory protection, protective clothing, and the use of radioprotective drugs by onsite emergency workers.

J. Protective Response

Standard

A range of protective actions has been developed for the plume exposure pathway FPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

Licensee Emergency Plan Evaluation

The applicant has described the protective actions to be taken by onsite personnel. Onsite predetermined assembly areas are designated. The station has an alarm system to signal personnel to assemble in these areas. Persons not having an emergency response assignment, including vistors and contractor personnel, are required to assemble when notified by the alarm. Onsite accountability is determined by the Station Security force.

Resolution of the Following is Needed

1. Explain the basis for adverse weather evacuation time estimates; and the alternative routes and methods of evacuation that will be used during inclement weather (e.g., snow, flood). In addition, the evacuation time estimates should be modified to consider institutions, such as correctional facilities, hospitals, nursing homes.

2. Describe the provisions for evacuation of nonessential personnel. Describe the evacuation routes, transportation, and decontamination capabilities.

3. Describe the use of radioprotective drugs by onsite personnel. Specify who decides when they are used, the dosage, and the amount available onsite.

4. Describe the recommendations for protective measures that may be given to the public based on measured or calculated dose rates specified in Appendix 1 of the criteria for each emergency condition.

APPENDIX G (Continued)

K. Radiological Exposure Control

Standard

Means for controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

Licensee Emergency Plan Evaluation

Emergency response personnel may receive radiation exposure in excess of the limits imposed by 10 CFR 20 when authorized by the Station Director. Emergency Plan Implementing Procedures and the Plan contain emergency guidelines for whole body and thyroid doses consistent with EPZ Emergency Worker and Life Saving Activity Protective Action Guides.

The station will provide and distribute self-reading and accumulative type dosimeters to personnel involved in emergency onsite response regardless of company affiliation. Dose records for workers will be maintained and checked daily throughout the emergency.

Onsite contamination control procedures for personnel, equipment, and access control are in place. Decontamination of personnel and equipment is required when the contaminaton level exceeds predetermined values. Criteria for permitting return of contaminated areas and their contents to normal use are stated in the appropriate contamination control procedures.

The staion will supply clothing and decontamination materials, particularly with respect to radioiodine skin contamination to onsite personnel required to relocate.

L. Medical and Public Health Support

Standard

Arrangements are made for medical services for contaminated injured individuals.

Licensee Emergency Plan Evaluation

The station emergency plan has provided for medical care at four facilities: (1) an onsite first aid facility, (2) the Christ Hospital, (3) the Bethesda Hospital, and (4) the Cincinnati General Hospital. Arrangements have been made with each of these offsite facilities. The purpose of these arrangements is to assure appropriate medical care.

Medical transportation is provided by onsite ambulance or the Moscow Life Squad.

Resolution of the Following is Needed

1. Only the agreement with Cincinati General Hospital is detailed. A letter of agreement with a backup facility should be formulated describing the facility's capabilities, personnel training, and the extent of treatment provided.

2. Describe how the injured will be transported during inclement weather from the station for medical treatment.

3. Specify where the Medical Director will be located during an emergency and how quickly he can arrive onsite.

M. Recovery and Reentry Planning and Postaccident Operations

Standard

General plans for recovery and reentry are developed.

Licensee Emergency Plan Evaluation

The applicant has described the mechanism used to progress from one emergency category to another. Also, procedures have been developed for reentry to previously evacuated areas for the purpose of saving lives, search and rescue of missing and injured persons, or manipulation, repair, or recovery of critical equipment or systems.

Resolution of the Following is Needed

1. Describe the means by which decisions are reached to relax both onsite and offsite protective measures.

2. Describe the means for informing members of the response organizations that a recovery operation is to be initiated, and of any changes in the organizational structure that may occur.

N. Exercises and Drills

Planning Objective

Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

Licensee Emergency Plan Evaluation

An emergency exercise will be conducted annually and will be based on an accident scenario which will simulate an emergency that results in offsite radiological releases and require response by offsite authorities. The scenario will be varied such that all plans and preparedness organizations are tested within a 5-year period.

Drills, which are supervised instruction periods aimed at testing, developing, and maintaining skills in the following areas, will be scheduled at the frequencies noted:

1. Communication drills - initial plant contact with State and county governments will be tested monthly; communications with Federal response agencies, offsite emergency centers, and field assessment teams will be tested annually.

2. Fire drills - according to Station Technical Specifications.

3. Medical emergency drills - annually.

4. Radiological monitoring drills - annually.

5. Onsite radiation protection drills - semiannually.

The Station Superintendent is responsible for the planning, scheduling, and coordinating of drills and exercises. All drills and exercises are approved by the Station Manager. The annual exercise is approved by the General Manager Nuclear Operations.

Each drill and exercise is conducted to test the state of emergency preparedness and is designed to meet a list of specific objectives which are specified in the plan. The Emergency Coordinator will coordinate and implement plan revisions and required corrective actions resulting from the drills and exercises.

Resolution of the Following is Needed

1. Change communications and fire drill frequency from annually to quarterly.

2. Include an annual medical exercise that includes participation by the licensee and local support services agencies.

3. Specify that scenarios will include:

    a. Arrangements for official observers

    b. The basic objective of the drill

    c. The date, time, place, and participating organization

    d. Simulated events

    e. The time schedule of real and simulated events

    f. A narrative summary description - but each scenario should allow "free play" for decisionmaking.

O. Radiological Emergency Response Training

Standard

Radiological emergency response training is provided to those who may be called upon to assist in an emergency.

Licensee Emergency Plan Evaluation

All personnel holding NRC licenses participate in a continuing requalification program. In addition, coordinators, managers, or supervisors assigned responsibilities and duty stations in the emergency organization receive annual refresher training. Emergency team members receive initial training and annual retraining. Personnel that receive Red Cross Multi-Media training are retrained every three years.

Resolution of the Following is Needed

1. Additional detail is required regarding the training program for personnel who will implement the radiological emergency response plan. The description should include the specialize training and periodic retraining programs (including scope, nature, and frequency) for each of the nine categories of personnel listed in Section II.O.4 of NUREG-0654.

2. The plan should indicate that formal training programs include training to determine individual qualifications, and any minimum levels of competence established for any of the (emergency response) positions. Training and retraining programs, qualifications testing, and competence should also include State and local officials.

3. Verify that first aid personnel will be retrained and tested annually.

4. Commit that offsite groups, such as fire departments, police and sheriff's departments, and ambulance services that may participate in onsite activity will be provided a training course to ensure that they are familiar with the plant layout and their role in the event of radiological and nonradiological incidents. Training of medical support personnel is discussed in the station medical plan.

P. Responsibility for the Planning Effort: Development, Periodic Review, and Distribution of Emergency Plans

Standard

Responsibilities for plan development, review, and distribution of emergency plans are established and that planners are properly trained.

Licensee Emergency Plan Evaluation

The Station Superintendent has overall authority and responsibility for radiological emergency response planning at the corporate level. The Rad/Chem Engineer implements the offsite aspects including arrangements and agreements with offsite organizations and personnel.

The Station Review Board is responsible for conducting a review of the Plan every twelve months.

The Operational Review Committee is responsible for independent audits every two years.

Resolution of the Following is Needed

Verify that independent audits will be conducted annually. In addition, items should include interfaces with State and local governments.

CONCLUSION

Based on our review, we conclude that the Zimmer Nuclear Power Station, upon satisfactory correction of the items identified in the preceding paragraphs, will meet the planning standards of 10 CFR 50.47(b), 10 CFR 50, Appendix E, and conform to guidance stated in NUREG-0654, Rev. 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," November 1980.

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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 discusses the resolution of these issues and the concerns of the Advisory Committee on Reactor Safeguards, which issued a favorable report on March 13, 1979. This supplement presents the Staff's bases for proceeding with licensing prior to resolution of the generic matters covered by the Task Action Plans outlined in NUREG-0410, "NRC Program for the Resolution of Generic Items Related to Nuclear Power Plants," and also presents the status of TMI Action Plan items listed in NUREG-0737, "Clarification of TMI Action Plan Requirements." The review will continue until the unit is operating. The Zimmer Nuclear Power Station is located in Washington Township, Clermont County, Ohio.

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