MAR 2 1982

DISTRIBUTION:
See attached page

Docket Nos. 50-440/441

RECEIVED
MAR 02 1982
US NUCLEAR REGULATORY COMMISSION
DOCUMENT MANAGEMENT BR
TIDC

Mr. Dalwyn R. Davidson
Vice President, Engineering
The Cleveland Electric Illuminating Company
P.O. Box 5000
Cleveland, Ohio 44101

Dear Mr. Davidson:

Subject: Draft SER: Perry Nuclear Power Plant (PNPP), Unit 1 and 2

Enclosed is a copy of the Draft SER for the PNPP, Units 1 and 2. The draft is incomplete and we will be transmitting the remaining portions as they become available.

Please review the draft carefully and provide responses to the open areas and questions contained therein. For the most part, these are highlighted and may be found under each section of the draft preceding the evaluation texts.

It is suggested that CEI be prepared to address the open areas and issues delineated in the draft. The NRC staff is prepared to meet with you in Bethesda, and/or have conference calls, to resolve the open areas, questions and issues. We want to encourage you to continue to have these types of meetings and discussions as they have been quite successful in expediting resolution of unresolved areas of the FSAR. We prefer that a draft response be prepared prior to holding any such meetings or discussions.

If you have any questions, please contact the Project Manager, Mr. John J. Stefano, at (301) 492-9536.

Sincerely,

Original Signed By:
F Miraglia

Robert L. Tedesco
Assistant Director for Licensing
Division of Licensing

8203100426 820302
PDR ADOCK 05000440
E PDR

Enclosure:
As stated

cc: See next page

OFFICE	DL:LB#2/PM	DL:LB#2/BC	DL:AD/L				
SURNAME	JStefano:kw	ASchwencer	RTedesco				
DATE	2/26/82	2/26/82	2/26/82				

Mr. Dalwyn R. Davidson
Vice President, Engineering
The Cleveland Electric Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

cc: Gerald Charnoff, Esq.
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N. W.
Washington, D. C. 20006

Donald H. Hauser, Esq.
The Cleveland Electric Illuminating Company
P. O. Box 5000
Cleveland, Ohio 44101

Resident Inspector's Office
U.S.N.R.C.
Parmly at Center Road
Perry, Ohio 44081

Donald T. Ezzone, Esq.
Assistant Prosecuting Attorney
105 Main Street
Lake County Administration Center
Painesville, Ohio 44077

Tod J. Kenney
228 South College, Apt. A
Bowling Green, Ohio 43402

Daniel D. Wilt
Wegman, Hesiler & Vanderberg
7301 Chippewa Road, Suite 102
Brecksville, Ohio 44141

Robert Alexander
OCRE Interim Representative
2030 Portsmouth Street
Suite 2
Houston Texas 77098

Terry Lodge, Esq.
915 Spitzer Building
Toledo, Ohio 43604

DISTRIBUTION LIST:

Docket File(s)
LB#2 File
NSIC
TERA
NRC PDR
L PDR
ACRS (16)
E.G. Case
D.G. Eisenhut/R. Purple
T. Novak
S. Varga
D. Vassallo
R.A. Clark
J.F. Stolz
R. Tedesco
B.J. Youngblood
A. Schwencer
F. Miraglia
E. Adensam
J.R. Miller
G.C. Lainas
T. Ippolito
W.T. Russell
D.M. Crutchfield
R.H. Vollmer
R.J. Mattson
H. Thompson
V. Moore
S.H. Hanauer
J.P. Knight
W.V. Johnston
D.R. Muller
P.S. Check
R.W. Houston
L.S. Rubenstein
F. Schroeder
M.L. Ernst
Attorney, OELD
OI&E
OSD (7)
R. Hartfield, MPA
Region III
Resident Inspector
J. Stefano
E. Hylton

NRC REVIEWERS

B.J. Elliot
J. Mauck
R. Capra
J. Kudrick
T. Greene
N. Fioravante
J.W. Clifford
H. Krug
G. Thomas
R. Giardina
M. Lamastra
P. Stoddart
W. Long
M. Hum
L. Yang
W. Brooks
S.C. Wu
C.E. McCracken
A. Brauner
F. Congel
D. Terao

bcc: Applicant Service List

DRAFT SAFETY EVALUATION REPORT

related to the operation of
PERRY NUCLEAR POWER PLANT
UNITS 1 and 2
Docket Nos. 50-440 and 50-441

CLEVELAND ELECTRIC ILLUMINATING COMPANY

U.S. NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

February 1982

"This draft SER has been assembled into one document to identify the open items from each review area, identified as of the date of issue of this draft, so that the NRC and The Cleveland Electric Illuminating Company staffs can work toward the objective of resolving them prior to issuance of the final SER. The Table of Contents conforms with NUREG-0800, and lists those NUREG-0800 sections for which draft evaluations are included in this draft SER. Where draft evaluations were not available at the time of assembly, the corresponding sections of NUREG-0800 have not been listed in the Table of Contents. The respective draft evaluations for the remaining sections of NUREG-0800 will be released as soon as they become available. In each of the sections included and not included in this draft SER, there is still potential for additional open items."

NUREG-0887

D R A F T

SAFETY EVALUTATION REPORT

related to the operation of
PERRY NUCLEAR POWER PLANT
UNITS 1 AND 2

DOCKET NOS. 50-440 and 50-441

CLEVELAND ELECTRIC ILLUMINATING COMPANY, et al.

U. S. NUCLEAR REGULATORY
COMMISSION

OFFICE OF NUCLEAR REACTOR REGULATION

FEBRUARY 1982

INDEX - TABLE OF CONTENTS

Section 2.0 Site Characteristics

2.1.1 Site Location and Description

2.1.2 Exclusion Area Authority and Control

2.1.3 Population Distribution

2.2 Nearby Industrial, Transportation and Military Facilities

2.2.1 Transportation Routes

2.2.2 Nearby Facilities

3.2 Classification of Structures, Systems and Components

3.2.1 Seismic Classification

3.3.3 System Quality Group Classification

3.4.1 Flood Protection

3.5.1.1 Internally Generated Missiles(Outside Containment)

3.5.1.2 Internally Generated Missiles(Inside Containment)

3.5.1.4 Missiles Generated by Natural Phenomena

3.5.2 Structures, Systems and Components to be Protected from Externally Generated Missiles

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Plant Design for Protection Against Postulated Piping Failure in Fluid Systems Outside Containment

3.6.2 Determination of Break Locations and, Dynamic Effects Associated with the Postulated Rupture of Piping

3.7.3 Seismic Subsystem Analysis

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.2 Dynamic Testing and Analysis of Systems, Components and Equipment

3.9.3 ASME Code Class 1,2 and 3 Components Supports and Core Support Structures

3.9.4 Control Rod Drive Systems

3.9.5 Reactor Pressure Vessel Internals

3.9.6 Inservice Testing of Pumps and Valves

4.3 Nuclear Design

4.3.1 Design Bases

4.3.2 Design Description

4.3.3 Analytical Methods

~~4.3.4~~
4.4 Thermal Hydraulic Design

4.4.1 Thermal Hydraulic Design Bases

4.4.2 Thermal-Hydraulic Design Methodology

4.4.2.1 Critical Power Ratio (CPR) Correlation

4.4.2.2 Thermal Hydraulic Stability

4.4.3 Design Abnormalacies

4.4.3.1 Crud Deposition

4.4.4 Loose Parts Monitoring

4.4.5 Thermal-Hydraulic Models

4.4.6 Thermal-Hydraulic Comparison

4.4.7 Single Loop Operation

4.6 Functional Design of Reactivity Control Systems

5.2.2 Overpressurization Protection

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

5.3.1 Reactor Vessel Materials
5.3.2 Reactor Coolant Pressure Boundary Materials

5.3.2 Pressure-Time Limits

5.3.3 Reactor Vessel Integrity

5.4.6 Reactor Core Isolation Cooling System

5.4.7 Residual Heat Removal System

5.4.8 Reactor Water Cleanup System

6.1.2 Organic Materials

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Containment Analysis

6.2.1.2 Review of Boiling Water Reactor Containment Technology

6.2.1.3 Short-Term Pressure Response

6.2.1.4 Long-Term Pressure Response

6.2.1.5 Containment External Pressure

6.2.1.6 Subcompartment Pressure Analysis

6.2.1.7 Steam Bypass of the Suppression Pool

6.2.1.8 Pool Dynamic Loads

6.2.1.8.1 LOCA Pool Dynamics

6.2.1.8.2 Relief Valve Dynamics

6.2.1.9 Secondary Containment

6.2.2 Containment Heat Removal System

6.2.3 Containment Isolation System

6.2.3.1 Containment Purge System

6.2.5 Combustible Gas Control

6.2.6 Containment Leakage Testing

6.2.7 TMI-2 Requirements

6.3 Emergency Core Cooling System

6.3.1 System Description

6.3.2 Evaluation

6.3.2.1 Single Failures

6.3.2.2 Qualification of the ECCS

6.3.2.3 Functional Design

6.3.3 Testing

6.3.3.1 Preoperational Tests

6.3.3.2 Periodic Component Tests

6.3.4 Performance Evaluation

6.3.5 Conclusions

6.4 Control Room Habitability

6.5.3 Fission Product Control Systems

6.5.1 ESF Atmosphere Cleanup Systems

6.7 Main Steam Isolation Valve Leakage Control System

9.0 Auxiliary Systems

9.1 Fuel Storage Facility

9.1.1 New Fuel Storage

9.1.2 Spent Fuel Storage

9.1.3 Spent Fuel Pool Cleanup System

9.1.4 Light Load Handling System (Related to Refueling)

9.1.5 Overhead Heavy Load Handling

9.2 Water Systems

9.2.1 Station Service Water System(Emergency Service Water System)

9.2.2 Reactor Auxiliary Cooling Water System

9.2.3 Demineralized Water Makeup System

9.2.4 Potable and Sanitary Water Systems

9.2.5 Ultimate Heat Sink

9.2.6 Condensate and Refueling Water Storage Facilities

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

9.3.2 Process Sample System

9.3.3 Equipment and Floor Drainage System

9.3.5 Standby Liquid Control System

9.4.1 Control Room Area Ventilation System
9.4.2 Spent Fuel Pool Area Ventilation System
9.4.3 Auxiliary and Radwaste Area Ventilation System
9.4.4 Turbine Area Ventilation System
9.4.5 Engineered Safety Feature Ventilation System

10.3.1 Main Steam Supply System (Up to and Including the Main Steam Block Valve)
10.4.2 Main Condensor Evacuation System
10.4.2.1 System Dewcription
10.4.2.2 Evaluation Findings
10.4.3 Turbine Gland Sealing System
10.4.3.1 System Descriptiom
10.4.5 Circulating Water System
10.4.6 Condensate Filter Demineralizer System
10.4.7 Condensate and Feedwater System

11.0 Radioactive Waste Management
11.1 Liquid Radioactive Treatment Systems
11.2 Gaseous Radioactive Treatment Systems
11.3 Solid Radioactive Waste Management Systems
11.4 Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems

12.1 Ensuring that Occupational Radiation Exposures are as Low is Reasonably Achievable (ALARA)
12.2 Radiation Sources
12.3 Radiation Protection Design Features
12.4 Dose Assessment
12.5 Health Physics Program

13.5.2 Operating and Maintenance Procedures
13.6.1 Plant Emergemcym Preparedness
Appendix 13A Draft Security Plan

15.0 Transient and Accident Analysis

15.1 Anticipated Operational Occurrences

15.2 Accidents

15.2.1 Anticipated Transients Without Scram

15.4 Rod Withdrawal Events

15.4.1 Inadvertant Rod Withdrawal at Low power

15.4.2 Rod Withdrawal Error at Power

15.4.7 Operation of a Fuel Assembly in an Improper Position-
Fuel Misloading Event

15.4.9 The Rod Drop Accident

15.X Radiological Consequences of Design Basis Accidents

15.X.1 Loss-of-Coolant Accident (Radiological Consequences)

15.X.2 Control Rod Drop Accident

15.X.3 Fuel Handling Accident

15.X.4 Steam Line Break Accident

15.X.5 Small Line Failure

22.0 TMI-2 Requirements

II.B.1 Reactor Coolant System Vents

II.B.2 Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Sysytems Which May Be Used In Postaccident Operations

II.F.1(3) Containment High-Range Radiation Monitor

II.K.1.22 Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems when Feedwater System is Not Operable

II.B.3 Post Accident Sampling Capability

II.K.3.13 Separation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation Levels

II.K.3.15 Modify Break Detection Logic to Prevent Spurious Isolation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System

II.K.3.44 Evaluationof Anticipated Transients with Single Failure to Verify No Fuel Failure

III.D.3.3 Improved Inplant Instrumentation Under Accident Conditions

INTRODUCTION & GENERAL DESCRIPTION
OF PLANT

SECTION 1

SITE CHARACTERISTICS

SECTION 2

SECTION 2

Status of Review
Geology and Seismology Input to the Safety
Evaluation Report
Perry Nuclear Power Plant, Units 1 and 2
The Cleveland Electric Illuminating Company
Docket Nos. 50-440 and 50-441

(OPEN AREAS FOR SECTIONS 2.5.1, 2.5.2 & 2.5.3)

Status of Geology-Seismology Input to the Safety Evaluation Report

The Geosciences Staff review of the Perry Nuclear Power Plant, Units 1 and 2 Final Safety Analysis Report is incomplete pending (1) staff evaluation of the Cleveland Electric Illuminating Company's (CEI) responses to nine of the twenty NRC and USGS questions and (2) applicant submission of responses to the eleven remaining NRC questions. The eleven responses are expected sometime during early December, 1981.

The two most significant issues to be resolved are (1) the determination of the capability or non-capability of the Cooling Water Tunnel faults and (2) a characterization of the SSE response spectra using either intensity or site specific relationships. Regarding fault capability both the USGS and the NRC have asked questions related to this issue. The applicant has responded to the USGS questions, but not the NRC's. CEI's responses to the seismicity issue have been received, but have not yet been thoroughly evaluated by the staff. The evaluation of these two significant issues as well as the others (see enumeration below) will be initiated shortly after the GSB's receipt of the eleven outstanding RAI responses. In addition to the two previously discussed matters, other items identified by the staff and requiring resolution include:

1. Tectonic Province - The applicant is proposing that the Grenville Front, located to the east of the Anna, Ohio area constitutes a tectonic province boundary in western Ohio, thus prohibiting the migration of that event to the site area in northeastern Ohio.
2. Updating the FSAR - The utility has been asked to provide assurance that both the geologic and seismologic information is current.

3. Response Spectra - CEI has been asked to compare the site response spectra to that suspected to have been generated in the past by earthquakes in the Cleveland area.
4. Offshore Geophysical Surveys - A number of geophysical surveys have been conducted in Lake Erie near the site. Consequently, the applicant has been asked to identify the various surveys and assess the value of the information derived as a basis for geologic structure definition.
5. OBE Exceedence Probability - The utility has been asked to discuss and provide the basis it used in describing the plant lifetime OBE exceedance probabilty.
6. Possible Geologic Structure - A number of anomalies (some based upon geophysical data; others upon well hole information), several within 5 miles of the site, have been identified in the FSAR. The applicant has suggested that a number of these anomalies may be fault controlled. The NRC has asked CEI to present its bases for identification of the anomalies.
7. Regional Faulting - Surface and subsurface faulting has been observed within 8 miles of the Perry site. The utility has been asked to discuss the faults in detail, bringing out the relationship of the regional faults to the Cooling Water Tunnel faults.
8. Lineament Analysis - CEI has been asked to conduct a lineament analysis of the area within 5 miles of the Perry plant.

Additional NRC questions may be asked as a result of our evaluations of (1) the applicant's recently-submitted nine responses and (2) the not-yet-received responses to the remaining eleven questions.

2.0 SITE CHARACTERISTICS

2.1 GEOGRAPHY AND DEMOGRAPHY

2.1.1 SITE LOCATION AND DESCRIPTION

The site for the Perry Nuclear Power Plant, which is a proposed two unit plant, consists of 1100 acres of land located in Lake County in northeast Ohio. Figure 2.1 shows the site location relative to the surrounding area. The site is about seven miles northeast of Painesville, the county seat, and 35 miles northeast of Cleveland, the largest city in the area. The site boundary and plant features are indicated on Figure 2.2. The plant exclusion area boundary is shown on Figure 2.3. Residential communities and other landmarks in the area within 10 miles of the site are illustrated in Figure 2.4. The coordinates of the Perry Unit 1 reactor are 41°48'4.2" north latitude and 81°8'36.6" west longitude. The Universal Transverse Mercator (UTM) coordinates are 4,627,498 meters north and 488,079 meters east, in zone 17. The site is in a rural area along the southeastern shoreline of Lake Erie about 50 feet above the low water datum, on an ancient lake plain which slopes gently toward the lake. Approximately 45% of the site land area is covered with woodland, the rest is used mostly for farmland and nursery stock.

2.1.2 EXCLUSION AREA AUTHORITY AND CONTROL

The applicant has defined the exclusion area for the Perry site as a circular area with a 2900 ft. (884 meters) radius, measured from the center of unit #1. The applicant owns all of the land and controls all of the mineral rights in the exclusion area, both on the land and within 1800 feet of all plant safety related structures located in Lake Erie. There are no residents living within the exclusion area.

There are no roadways or railroads traversing the exclusion area. Activities within the exclusion area unrelated to plant (unit 1) operations are limited primarily to activity associated with the construction of unit 2 and the water related activities on nearby Lake Erie. Formal arrangements have been made with the U.S. Coast Guard, in case of emergencies, to control that portion of the exclusion area which extends into Lake Erie. Refer to Section 13.3 of this report for more details of these arrangments.

We conclude, by virtue of ownership of the land and control of the mineral rights within the exclusion area, and on the basis that suitable arrangments have been made to control all activity on that portion of Lake Erie within the exclusion area, that the applicant has the authority to determine all activities within the exclusion area, as required by 10 CFR Part 100. We also conclude that activities unrelated to plant operation within the exclusion area will not interfere with normal plant operation.

2.1.3 POPULATION DISTRIBUTION

The resident population in the vicinity of the Perry site is shown as a function of distance in the table below. 1983 is the estimated year of plant startup, while 2020 is the nearest census year to the end of plant life.

TABLE 2.1 RESIDENT POPULATION VS. DISTANCE

Year	0-1 Miles	0-2 Miles	0-3 Miles	0-4 Miles	0-5 Miles	0-10 Miles
1978	103	1818	5723	10643	16875	73134
1983	103	1818	5725	10648	16885	74085
2020	115	2043	6431	11958	18959	86443

The closest residence is about 3200 ft. (975 meters) from the reactor. The nearest community in the vicinity of the site is North Perry, located 1.5 miles southwest of the plant with a 1980 population of 896. The closest large community nearby is Painesville, about seven miles southwest of the plant with a population of 16,351 in 1980. The population within five miles of the site in 1978 was 16875 and within ten miles it was 73134. As can be seen in Table 2.1 the population within five miles is only expected to increase by ten persons in the five year period between 1978 and 1983, and by 2084 persons during plant life (2020). The applicant reported that there was 2,480,678 people living within 50 miles of the Perry site in 1978. In 1980 this figure dropped to 2,451,640, and it is expected to decrease to 2,435,526 by 1983. The applicant expects the population in this area to continue to decline until about 1990 when they predict a population of 2,401,526. By the year 2000 it is projected that the population will start to increase again and in the year 2020 it is expected to reach 2, 413,435 which is still below the 1978 population figure within 50 miles. The closest major city within 50 miles of the site is Cleveland, located 33 miles southwest, with a 1980 population of 572,657. The applicant predicted a negative population growth rate for the area within 50 miles of the site during the life of the plant as compared to a 0.48%/year growth predicted by the Bureau of Economic Analysis for this area.

The applicant has selected a low population zone for the site with a radius of 2.5 miles (4023 meters), measured from a point midway between units 1 and 2. As with most of the land surrounding the Perry plant, some portions of the site property within the LPZ are farmland, ~~having~~ forested areas, and nurseries. A small fraction of this property managed by private individuals will still be productive after unit #1 goes into operation. ~~but will be under the control of the applicant.~~ There were about 4225 residents living within the LPZ in 1978. This figure is not expected to vary appreciably during the life of the plant. The peak transient population in the LPZ in 1979 was approximately 1575/day during the school year. Of these, 175 are employed by the Neff-Perkins Company which is located 3000 feet from the plant. The remaining 1400 persons attend three schools, each 2.2 miles from the plant. The contribution ~~from~~ from the various parks and camps in the LPZ is small. There are no other institutions within the LPZ. Section 13.3 of this report provides a discussion of the Emergency Preparedness Plans for protection of the public in the area.

The applicant has indicated that the nearest densely populated center, as defined by 10 CFR Part 100, of about 25,000 or more persons is Painesville, Ohio. The population of Painesville was 17, 407 in 1975 and 16,351 persons in 1980. However, the combined population of Painesville and the communities immediately surrounding it is expected to exceed the 25,000 person criteria before the end of plant life and thereby become the nearest densely populated center. The distance from the closest corporate boundary of Painesville (6 miles) to the site is at least one and one-third times the distance to the LPZ outer radius, as required by 10 CFR Part 100.

2.1.4 CONCLUSION

On the basis of the 10 CFR Part 100 definitions of the exclusion area, the low population zone and the population center distance, our analysis of the onsite meteorological data from which the relative concentration factors (X/Q) were calculated (See Section 2.3 of this report), and the calculated potential radiological dose consequences of design basis accidents (See Section 15.0 of this report), we have concluded that the exclusion area, low population zone and population center distance meet the criteria of 10 CFR Part 100 and are acceptable.

2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

2.2.1 TRANSPORTATION ROUTES

There are no roads traversing the Perry exclusion area. Lockwood and Center Roads, which formerly ran through the exclusion area have been withdrawn from public use. Lockwood Road terminates at the site boundary and Center Road terminates at the exclusion area and is now used as the plant access road inside the exclusion area. Several highways pass within five miles of the Perry plant. The closest highway, U.S. Route 20, is located one mile south-southeast of the plant. State Route 84 and Interstate 90 run parallel to U.S. 20 at distances of 3.5 and 5 miles, respectively. State Route 2 joins U.S. 20 in an easterly direction about four miles southwest of the plant, and State Route 528 which runs north and south, intersects each of the above mentioned highways approximately five miles east of the pla.... The applicant has performed a survey of the traffic and hazardous materials transported on all roads within five miles of the plant. An analysis on the effects of potential explosive hazards and toxic material releases transported on the highways in the area was performed. The applicant concludes that the separation distance as well as the nature of the roads in the vicinity of the plant is sufficient to preclude adverse effects on the plant in the event of such explosions or releases. The staff, after reviewing the applicants data, concurs.

There are three railroad lines in the vicinity of the plant, the Norfolk and Western (N&W), Consolidated Rail Corporation (Conrail), and the Fairport, Painesville and Eastern (FP&E). The FP&E is a local line responsible for operations on the CEI owned rail spur which serves the Perry site. Conrail and N&W which are larger systems, operate on longer distance routes. These three rail lines run across northern Ohio and converge in a major corridor about three miles south of the plant at its closest point. These railroads transport practically all types of hazardous materials that are legally transported. The applicant has evaluated the potential consequences of a hypothetical propane explosion, and the release of toxic materials such as chlorine and ethylene oxide. The Perry plant is equipped with detectors which will automatically isolate the control room in the event of a chlorine or ethylene oxide release, and based on the distance involved, the applicant concludes that there will be no adverse effects on the plant from accidental explosions on the railroads. The staff concurs with this analysis.

Lake Erie is the only navigable waterway in the vicinity of the Perry site. The nearest shipping channel in Lake Erie is about two miles ~~of the site.~~ from the plant. The applicant has analyzed the effects of an explosion on the closest safety related structures at the plant and has concluded that the distance, almost two miles from the shipping channel, precludes any adverse effects on plant operations. The potential loss of cooling water as a result of waterborne collisions involving the intake and discharge structures was also considered. These structures, which are separated by at least one-third of a mile, are located in approximately 20 feet of water within one-half mile of the shoreline and about 1.8 miles from the nearest shipping channel. These structures are designed with additional physical protection to minimize the consequences of impact so that the probability of flow blockage in the intake and discharge structures is extremely remote. Based on the separation distance and design features of the cooling structures we conclude that activities on this portion of Lake Erie will not effect the operation of the plant.

Based on the nature of the transportation routes, the separation distances involved and previous staff review experience, we conclude that accidents associated with nearby transportation routes will not present a hazard to the safe operation of Perry Unit 1.

2.2.2 NEARBY FACILITIES

There are no military bases or missile installations within 10 miles of the Perry site. There are no airports within five miles of the plant. A small sod airstrip is located about 4.5 miles east-southeast of the plant which bases four single engine planes with only one operation per week. Within 10 miles there are two airports with one paved runway each. Casement airport, located six miles south-southwest, bas. 40 single and twin engine aircraft and has about 15 to 20 flights/day. Concord airport, located 10 miles south-southwest, bases 39 single engine and 3 twin engine aircraft but does not keep any operations records. Another airport, Lost Nation, is located 15 miles southwest, bases 150 aircraft and has almost 70,000 flights/year. The nearest major airport with commercial facilities is Hopkins International Airport about 45 miles southwest of the site.

There are no low level military aircraft training routes near the site. There are four low altitude (V10, V188, V188-10 and V14-W), and two high altitude (J584 and J29-82) airways within a 10 mile radius of the site. At its closest point, the centerline of V10 and V188 are each located about 1.5 miles southwest and west-northwest, respectively. V188-10 is 3.5 miles north-northwest and V14-W is 6.5 miles north of the plant. The centerlines of J584 and J29-82 are four miles north-northeast and 9.5 miles south-southeast of the plant, respectively.

The applicants assessment of aircraft hazards at the site has been independently verified by the staff. We have concluded that the probability of an aircraft crash causing radiological consequences in excess of the guidelines of 10 CFR Part 100 are within the acceptance criteria of Standard Review Plan Section 2.2.3 (less than about 10^{-7} per year), and is, therefore, acceptable.

There are five industrial firms within five miles of the plant. The NEff-Perkins Company is located 3000 feet west-southwest of the plant. Perry Coal & ~~Feu~~ Fuel Company, Calhio Chemicals, Inc., and the A. A. Covell Company are all south of the plant 3, 3.5 and 4.3 miles, respectively. Glyco Chemicals is five miles west-southwest of the plant. The applicant has performed an analysis of the various chemicals or hazardous materials which are stored and/or used at the different facilities, and has concluded that because of the distances involved from these facilities to the site and the quantities of material available, they do not present a hazard to the plant. Butadiene, which is stored at the Calhio Chemical Company (3.5 miles) was considered as being a potential threat to the plant, but according to the criteria of Regulatory Guide 1.78 this material does not pose a problem. We concur with this analysis. There is no appreciable change or expansion of industrial facilities anticipated in the immediate vicinity of the plant in the foreseeable future.

There are 13 pipelines within the site environs. These are all gas pipelines which vary in size from 1.25" to 20" with operating pressures ranging from 35 psi to 150 psi. As a result of the staff's concern at the construction permit stage, the 20" ~~gaso~~ pipe line (which was initially a 16" line) has since been upgraded and relocated to its present position. The applicant performed an analysis of each of the 13 pipelines to determine the limiting potential accident conditions. The analysis indicated the closest pipeline, four inches in diameter operating at 35 psi and 3000 feet southwest of the plant, and the largest pipeline, 20 inches in diameter operating at 150 psi and 3200 feet southeast of the plant were the two lines presenting the greatest potential threat to operations at the plant. The analysis was based on potential accidents involving the release of gas and the detonation of unconfined gas-air mixtures. The applicants analysis indicates that detonation from the unconfined gas-air mixture is not considered to be a credible event and that concentrations at the plant air intakes from a gas leak would be below the lower flammable limit and would not affect the safety of the plant. The staff concurs in this.

The staff also had some concern during the CP stage about the underground gas storage facilities in the vicinity of the plant, particularly propane storage. The staff informed the applicant (SER-CP and subsequent Amendments) that we would require the applicant to establish a separation distance of one mile between any potential underground storage release point and the nearest safety-related structures of the plant. CEI has responded (FSAR) by obtaining all the propane storage rights within at least one mile of the plant which will maintain an adequate separation distance between a potential release point and the nearest safety related structures. We conclude that this action will limit any adverse consequences to the Perry plant resulting from a postulated well-head release.

There are several active oil and gas producing wells within five miles of the plant. The closest gas well is located one mile northeast, and the closest oil well is 1.5 miles west-southwest of the plant. In our July 17, 1981 letter to CEI, the staff requested additional information in order to complete its review of the operating license application. On September 11, 1981, CEI responded to all the requests except the one pertaining to the oil and gas wells, and stated that this reply would be forthcoming by mid-November. We have not received CEI's response at this time and, therefore, cannot make a finding on this matter. We expect this item to be satisfactorily resolved prior to plant operation, and will report the results of our evaluation in the final safety evaluation report.

2.2.3 CONCLUSIONS

On the basis of the information provided by the applicant, and our review based upon criteria in 10 CFR Part 50, Appendix A, GDC4, and in Standard Review Plan Section 2.2.3, we have determined, subject to the satisfactory evaluation of the gas and oil well analysis by CEI, that the Perry plant is adequately protected and can be operated with an acceptable degree of safety considering the activities at nearby transportation, industrial, and military facilities.

Figure 2.1
Site Location
Relative
To Surrounding
Area

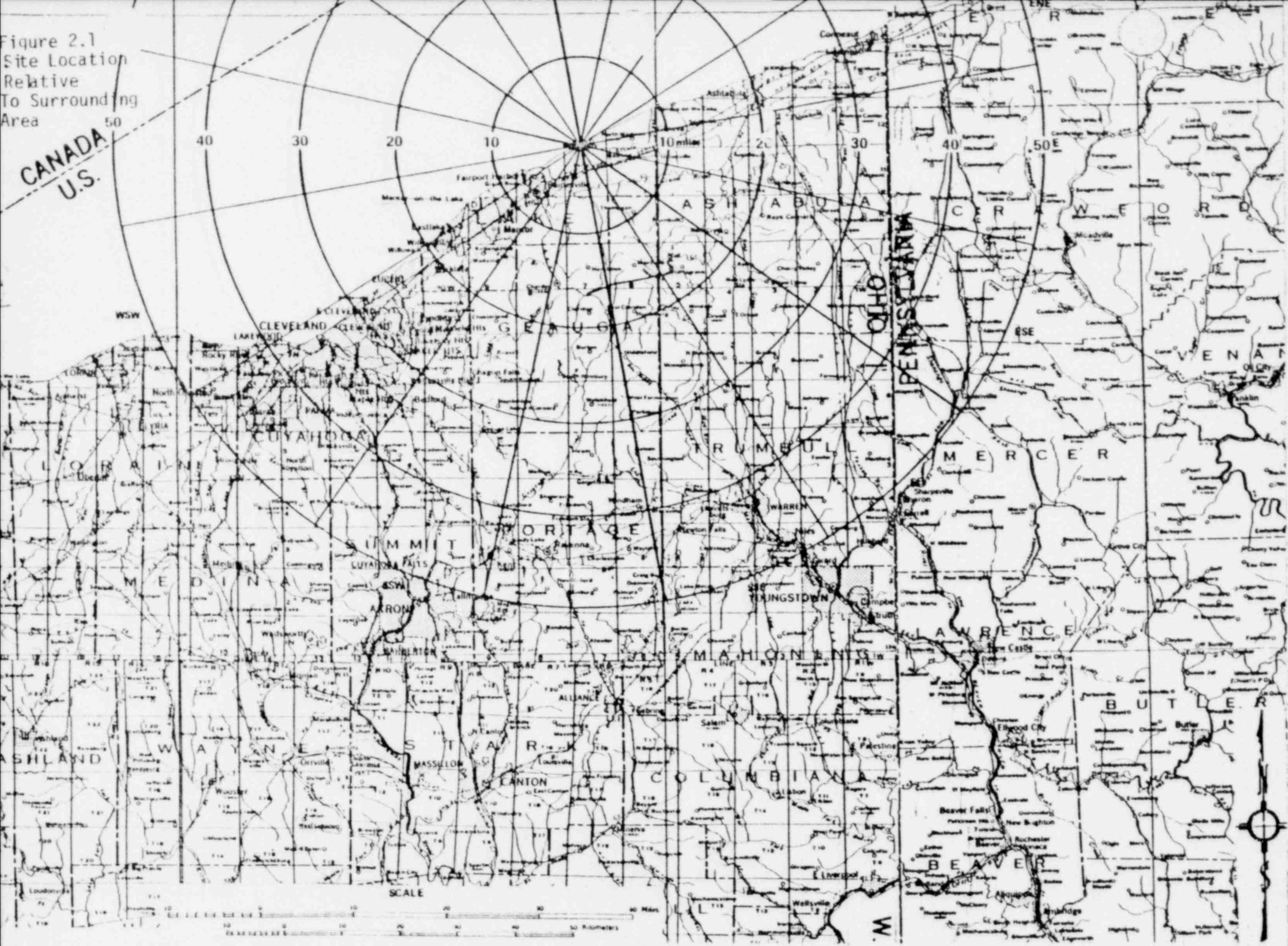


Figure 2.2
Site Boundary
And Plant Features

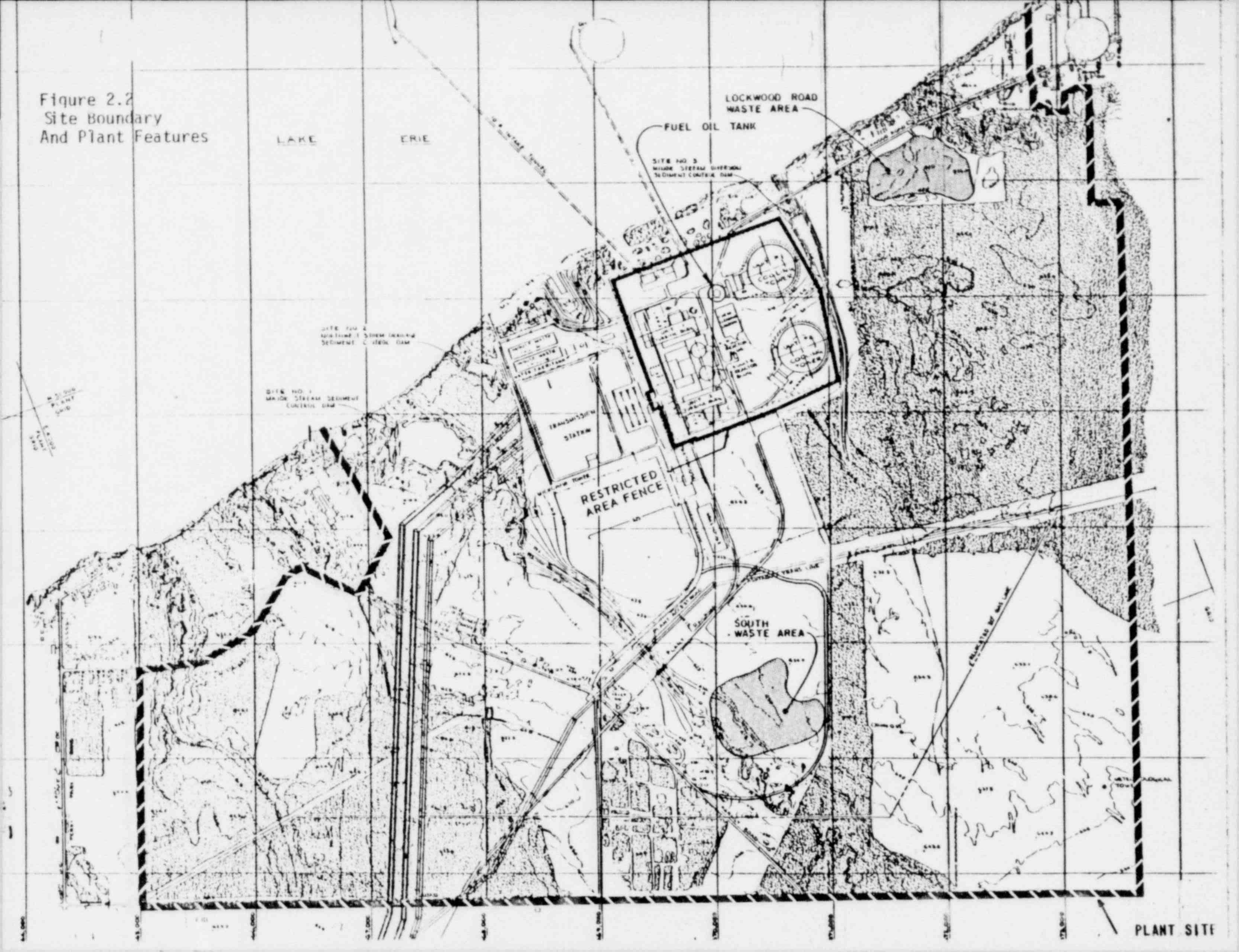


Figure 2.3
Exclusion Area Boundary

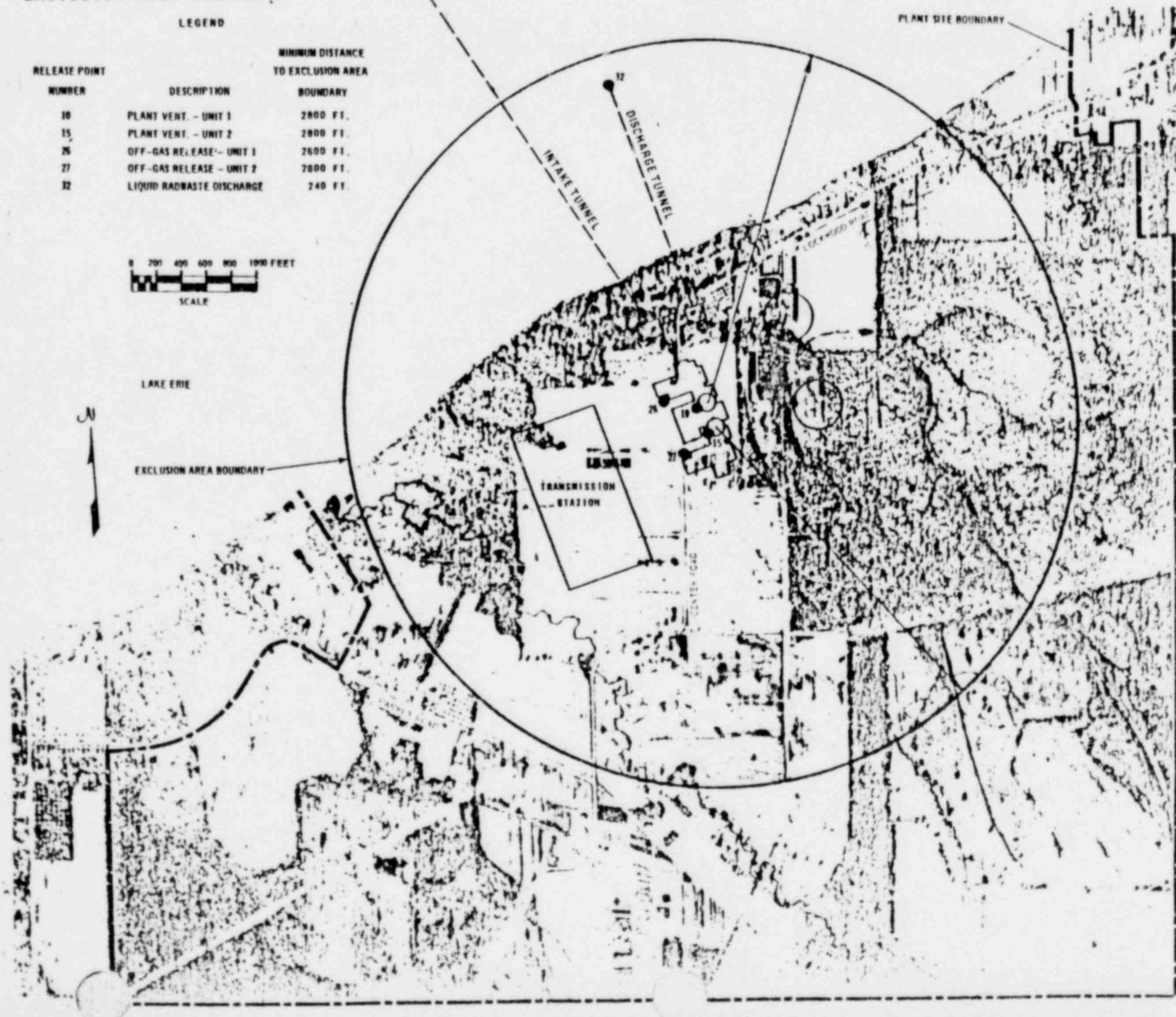
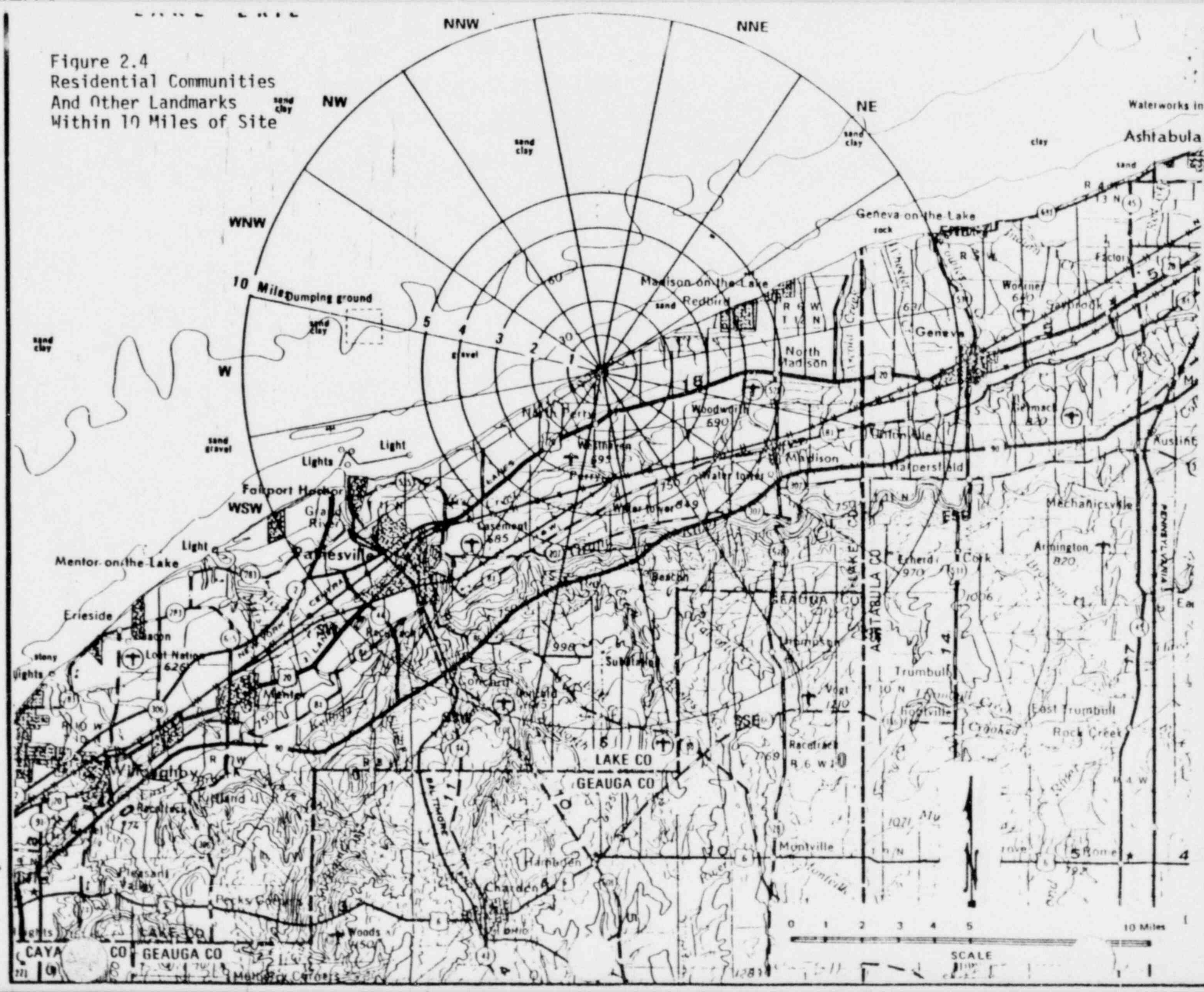


Figure 2.4
Residential Communities
And Other Landmarks
Within 10 Miles of Site



DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

SECTION 3

SECTION 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

OPEN AREAS

SECTION 3.4.1 - FLOOD PROTECTION - We have not completed our review of THE APPLICANTS RESPONSE to question 410.5 concerning internal flooding. No action required by the applicant.

Questions unacceptably or incompletely responded to by CEI (ltr of 1/6/82)

QUESTIONS ON PERRY FSAR

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

3.2.1 Seismic Classification

3.2.1, Page 3.2-1

It states in the FSAR that structures, components and systems designated as Safety Class 1, 2, or 3 are classified as Seismic Category I except for some portions of the radioactive waste treatment handling and disposal systems. There are several items in Table 3.2-1 in conflict with this statement.

3.2.1, Page 3.2-2

"The seismic classification indicated in Table 3.2-1 meets the requirements of Regulatory Guide 1.29." It is also stated in Section 1.8 that the Perry plant complies with all the requirements of Regulatory Guide 1.29. Does this mean that seismic Category I cooling water is provided to the recirculation pump during normal operation and following LOCA?

Table 3.2-1, Page 3.2-9

Quality assurance requirements should be addressed in this table.

Table 3.2-1, Page 3.2-9

What design requirements were used in the design of the reactor pressure vessel skirt?

Table 3.2-1, Page 3.2-9

Justify the non-seismic classification of the control rods. Note 7 does not apply to the control rods.

Table 3.2-1, Page 3.2-9

Provide an explanation for the "I, NA" seismic classification for relief valve discharge piping.

Table 3.2-1, Page 3.2-10

How much of the main steam piping, between the M.O. stop valve and the turbine stop valve, is located in the Auxillary Building?

Table 3.2-1, Page 3.2-24

There appears to be a discrepancy in the seismic classification of the discharge tunnel. The discharge tunnel and the diffusor nozzle are seismic Category I. The tunnel entrance structure and downshaft are not. Provide clarification for this apparent contradiction.

Table 3.2-1, Page 3.2-25

What is the seismic classification of the Containment Vessel Cooling Units?

Table 3.2-1, Page 3.2-34

Note 19 is an exception to Regulatory Guide 1.29 and should be included in Section 1.8.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.2, Page 3.6-7

In Section 3.6.1 references are made to "elastic/plastic pipe whip restraints or pipe supports which eliminate pipe whip damage". Details of how pipe supports are designed for pipe whip protection and an example of such an analysis are needed.

3.6.2.1.4, Page 3.6-10

How is it determined that "The internal energy associated with whipping is insufficient to impair the safety function of any structure, system or component to an unacceptable level"?

~~3.6.2.1.5, Page 3.6-10~~

~~The definition of terminal ends should be expanded to show that rotational movement from dynamic or static loadings are also restrained.~~

3.6.2.1.5, Page 3.6-11

Plant loading conditions for evaluating pipe break are to include normal and upset conditions plus an OBE. Assurance must be provided that SRV discharge loads are included in the upset conditions.

3.6.2.1.5, Page 3.6-11

For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks due to stress are to be postulated at the following locations:

-4-

(1) If Eq. (10), as calculated by Paragraph NB-3653, ASME Code Section III, exceeds 2.4 S_m, then Eqs. (12) and (13) must be evaluated. If either Eq. (12) or (13) exceeds 2.4 S_m, a break must be postulated. In other words, a break is postulated if

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} > 2.4\ S_m$$

or

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (13)} > 2.4\ S_m$$

(2) Breaks must also be postulated at any location where the cumulative usage factor exceeds 0.1.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

Any deviations from the above criteria must be justified.

3.6.2.1.5, Page 3.6-11

Are there any high energy Class 2, Class 3, or B31.1 lines? If so, what criteria is used for postulating breaks in these lines?

3.6.2.1.6, Page 3.6-13

Any instances where longitudinal break areas are less than one circumferential pipe area must be identified. The analytical methods representing test results and based on a mechanistic approach must be explained or justified. Provide examples of a typical analysis.

3.6.2.1.6, Page 3.6-14

How are energy reservoirs of sufficient capacity to develop a jet flow determined? What are justifiable line restrictions? Provide the justification. Any instances where flow limiters are used should be identified and justified.

3.6.2.1.7.1, Page 3.6-15

For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks need not be postulated providing the following stress criteria is met.

(1) If Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III does not esceed 2.4 S_m, a break need not be postulated.

(2) If Eq. (10) does exceed 2.4 S_m, Then Eqs. (12) and (13) must be evaluated. If neither Eq. (12) or (13) exceeds 2.4 S_m, a break need not be postulated. In other words, a break need not be postulated if

$$\text{Eq. (10)} < 2.4\ S_m$$

or

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} < 2.4\ S_m$$
$$\text{and Eq. (13)} < 2.4\ S_m$$

(3) Breaks need not be postulated as long as the cumulative fatigue usage factor is less than 0.1.

(4) For plants with isolation valves inside containment, the maximum stress, as calculated by Eq. (9) in ASME Code Section III, Paragraph NB-3652 under the loadings of internal pressure, dead weight and a postulated piping failure of fluid systems upstream or downstream of the containment penetration area must not exceed 2.25 S_m.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

In addition, augmented inservice inspection is required on all piping in the break exclusion area.

The applicant must provide assurances that their criteria for piping in the break exclusion areas complies with the requirements outlined above and those of Standard Review Plan 3.6.2.

3.6.2.1.7.1, Page 3.6-15

Are there any Class 2, Class 3 or B31.1 piping in the break exclusion areas? If so, what criteria is used for their design?

3.6.2.1.7.1, Page 3.6-15

A list of all systems in the break exclusion area is needed. Break exclusion area should be shown on the appropriate piping drawings.

3.6.2.1.7.2, Page 3.6-15

Provide an example of the detailed stress analysis done on a welded attachment to the process pipe. In addition, provide details of the stress analysis done on the head fitting for the main steam line.

3.6.2.2.1, Page 3.6-17

Provide a list of all locations where limited break opening areas have been used. Provide justification for each location and details of any inelastic analysis used.

3.6.2.2.1, Page 3.6-17

Provide a list of all locations where break opening times greater than one millisecond have been used. Provide and justify any experimental data and analytical theory.

3.6.2.2.2, Page 3.6-20

Provide assurance that all potential targets are evaluated when considering pipe whip.

3.6.2.2.2, Page 3.6-20

Provide a definition for limits of strain which are similar to strain levels allowed in restraint plastic members.

3.6.2.2.2, Page 3.6-20

"Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show the consequences do not result in direct damage to any essential system or component." Provide a list of where this technique has been used and an example of the studies performed.

~~3.6.2.3, Page 3.6-22~~

~~Provide details of the structural design and analysis of the steam tunnel as stated in Section 3.6.1.2.2.~~

3.6.2.3.1, Page 3.6-23

It is the staff's position that when evaluating jet impingement loads <u>all</u> potential targets must be evaluated. Provide assurances that your analysis for jet impingement effects have included all possible targets.

3.6.2.3.1, Page 3.6-29

What service limits are used for piping when evaluating jet impingement loads?

3.6.2.3.1, Page 3.6-30

How is it determined that the dynamic load factor (DLF) is suitable? Provide an example of its use.

3.6.2.3.1, Page 3.6-30

For snubbers, what are the "other simultaneous loads" that are combined by the SRSS method?

3.6.2.3.3, Page 3.6-33

"Piping integrity usually does not depend upon the pipe whip restraints for any loading combination." List all those locations where integrity of the piping depends upon the pipe whip restraints.

3.6.2.3.3, Page 3.6-33

What service limits are used in the design of the pipe whip restraints?

3.6.2.3.3.1, Page 3.6-33

What critical locations inside containment are monitored during hot functional testing?

3.6.2.3.3.1, Page 3.6-40

Any locations where the increase in the yield or ultimate strengths, of the material used for pipe whip restraints, exceeds 10% must be identified. Justification for any increase greater than 10% must also be provided.

3.6.2, Tables

Provide a schedule for the completion of any table that is incomplete.

3.6.2, Figure 3.6-66

Are all postulated break locations in the recirculation system shown?

3.6.2, Figures 3.6-71, 3.6-73, 7.3-74, 3.6-77, 3.6-78, 3.6-79, 3.6-80

Where are breaks postulated in these figures?

3.6.2, Figure 3.6-75

Indicate the location of valves in this line.

3.7.3 Seismic Subsystem Analysis

3.7.2.1.2.5, Page 3.7-11

The discussion on "Different Seismic Movement of Interconnected Components" requires some clarification. "The stresses thus obtained for each natural mode are then superimposed for all modal displacements of the structure by the SRSS (square root sum of the squares) method." Provide an example of what was done here.

3.7.2.1.2.5, Page 3.7-11

What criteria was used to determine whether or not a mode was significant?

3.7.2.1.2.5, Page 3.7-11

"When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses." Does this statement pertain to piping or supports?

3.7.3.1.1, Page 3.7-20

"Seismic analyses were performed for those subsystems that could be modeled to correctly predict the seismic response." What procedure was used for the other systems? Provide an example of some of those systems.

3.7.3.1.1, Page 3.7-21

What is meant by "Closely spaced in phase modes"?

3.7.3.2.1, Page 3.7-21

How many stress cycles are used in the BOP design?

~~3.7.3.3.2.1, Page 3.7-23~~

~~"If the torsional effect is expected to cause pipe stresses less than 500 psi, this effect may be neglected". How are these stresses estimated? Justify the 500 psi limit~~.

3.7.3.3.2.1, Page 3.7-23

Part (a) discussing decoupling of main steam and branch lines is not a criteria.

3.7.3.3.2.2, Page 3.7-24

Mention is made of using 33 hertz as a frequency cutoff for seismic analysis. At some point in the FSAR the applicant must address the frequencies of 50 to 60 hertz and greater than come from the suppression pool hydrodynamics.

3.7.3.5, Page 3.7-25

"For flexible equipment, the equivalent static load is taken as the product of 1.5 times the equipment mass and the peak floor response spectrum value." Regulatory Guide 1.100 allows the use of the 1.5 factor for verifying the integrity of frame type structures. For equipment having configurations other than a frame type structure, justification is required for use of the 1.5 factor.

3.7.3.7.1, Page 3.7-26

What procedure is used for combining closely spaced modes of systems in the BOP scope?

3.7.3.7.2, Page 3.7-26

The referenced equation should be as follows

$$R = \left[\sum_{K=1}^{N} \sum_{S=1}^{N} \left| R_K R_S \right| \epsilon_{KS} \right]^{1/2}$$

3.7.3.8.1, Page 3.7-28

Justification must be provided that the modeling of valves with off-set motor operators is detailed enough to provide acceleration values to be used for valve qualification.

3.7.3.8.1, Page 3.7-28

"In addition, the effects of the modes not included are added to the SRSS response as one term, using the acceleration at the highest frequency from the SRSS response under 33 hertz to obtain the total response." Provide an example of what was done here.

Table 3.7-11, Page 3.7-54

Provide a detailed explanation of the information in this table.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9, Page 3.9-1

Any references to the ASME Boiler and Pressure Vessel Code should indicate what part is being referenced.

3.9.1.2, Page 3.9-1

Methods of verification are required for all NSSS computer codes used in the analysis.

3.9.1.2.6, Page 3.9-16

All computer programs used in the design and analysis of systems and components within the BOP scope must be listed. Methods of verification are required for all BOP programs.

3.9.1.4.12, Page 3.9-26

It is stated that elastic-plastic methods of analysis may be used for some components. We would like to review the analysis procedures that would be used if an elastic-plastic analysis was done.

3.9.2, Page 3.9-27

More detail is needed for the NSSS and BOP preoperational vibration testing program. What locations will be monitored. What types of instrumentation will be used. What are the actual values that will be used for deflection and stress limits.

The staff's position is that acceptance limits for vibration should be based on half of the endurance limit as defined by the ASME Code at 10^6 cycles. We will require a copy of your results from your preoperational vibration testing program.

3.9.2.1.2, Page 3.9-29

"The piping system does 'shakedown' after a few thermal expansion cycles." Provide an explanation of this statement.

3.9.2.4, Page 3.9-65

"In addition to the above components, vibration measurements of the core spray sparger will be measured during preoperational testing of that system at the designated prototype 251 BWR/6 plant (Grand Gulf)." Show how this is applicable to Perry.

3.9.2.4.1, Page 3.9-66

Provide a commitment that Perry will be in compliance with Regulatory Guide 1.20 for prototype reactors.

3.9.2.5, Page 3.9-67

"These periods will be determined from a comprehensive dynamic model of the RPV and internals with 12 degrees of freedom." It is not clear what is actually done here. How can a model be comprehensive and have only 12 degrees of freedom?

3.9.2.6, Page 3.9-68

It appears that some results from Grand Gulf will be used in the evaluation and qualification of the reactor internals at Perry. Show that the similarity between the two sets of internals is sufficient to allow direct comparisons.

3.9.3, Page 3.9-68

Several references are made throughout this section to allowable stresses for bolting. Specifically, what allowable stress limits are used for bolting for (a) equipment anchorage, (b) component supports, and (c) flanged connections? Where are these limits defined?

3.9.3.1.2, Page 3.9-78

Are there any Class 1 systems in the BOP scope of responsibility?

3.9.3.4.1, Page 3.9-107

"For the NSSS scope of supply, all valve operators which are mounted on Class 1 piping will not be used as attachment points for component supports." What about Class 2 and 3 piping? This question also applies to the BOP scope of responsibility.

3.9.3.4.1, Page 3.9-109

Provide more detail on the testing done on snubbers.

3.9.3.4.4, Page 3.9-112

What elastic-plastic analysis has been done on supports? Provide an example of this analysis.

3.9.4.3, Page 3.9-114

Reference is made to allowable deformation in the title of this section but there is no discussion of allowable deformations in the text.

3.9.5.1.1.8, Page 3.9-120

Recently, cracking has been observed in BWR jet pump holddown beams. The resolution of this problem may affect the design or testing of the Perry jet pumps (see I&E Bulletin 80-07).

3.9.5.1.1.10, Page 3.9-121

What feedwater sparger design and Control Rod Drive Return Line modifications is used at Perry? Provide a commitment to NUREG-0619.

3.9.5.3.3, Page 3.9-129

Have the reactor internals placed in the "other internals" category been seismically analyzed to show that they will not compromise the integrity of seismically qualified reactor internals?

3.9.6, Page 3.9-131

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to pretect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak-tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an intersystem LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specification which will require corrective action; i.e., shutdown or system isolation when the final approved leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting valve would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

Table 3.9-1, Page 3.9-134

Does this table apply to Perry?

Table 3.9-1, Page 3.9-135

What does "1*****" refer to?

Table 3.9-1, Page 3.9-135

How many ADS cycles are included in the design of Perry?

Table 3.9-1, Page 3.9-136

Standard Review Plan 3.9 requires 5 OBEs of 10 cycles each. If fewer cycles are used, justification must be provided.

Table 3.9-3, Page 3.9-141

The acceptance criteria should reference the ASME Code Service Limits. A similar table is needed for the BOP.

Table 3.9-3a, Page 3.9-143

"The results of stress and fatigue usage analysis are given in detail in the vessel manufacturer's stress report and in new loads evaluation by GE within the code limits." Provide clarification of this statement.

Table 3.9.3m, 3.9.3o, 3.9.3q and 3.9.3h

Some values in these tables are missing. Provide a schedule for their completion.

Table 3.9-3s, Page 3.9-225

Provide an explanation for the results in this table.

Table 3.9-28, Page 3.9-282

Where are the loads used in this table defined? How are these loads combined?

Table 3.9-32, Page 3.9-297

Has Eq. b) been used? If so, provide the supporting data. If not, delete the equation from the table.

Table 3.9-33, Page 3.9-298

Have Eqs. e), f), or g) been used? If so, provide the supporting data. If not, delete these equations from the table.

Table 3.9.34, Page 3.9-301

Has Eq. c) been used. If so, provide the supporting data. If not, delete the equation from the table.

ADDITIONAL QUESTIONS

Table 3.2-1, Page 3.2-9

What design requirements were used in the design of the core support structures?

3.6.2.1.6, Page 3.6-13

Regardless of the ratio of longitudinal to hoop stress, both a longitudinal split and a circumferential break should be postulated at any location where the cumulative usage factor is greater than 0.1.

3.9.1.1.1, Page 3.9-1

How many cycles due to SRV discharge are included in the analysis?

3.9.2.5, Page 3.9-67

Previous analyses for other nuclear plants have shown that certain reactor system components and their supports may be subjected to previously under-estimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations.

The applicant has described the design of the reactor internals for blowdown loads only. The applicant should also provide information on asymmetric loads. It is, therefore, necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. The reactor system components that require reassessment shall include:

a. Reactor pressure vessel
b. Core supports and other reactor internals
c. Control rod drives
d. ECCS piping that is attached to the primary coolant piping
e. Primary coolant piping
f. Reactor vessel supports

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above mentioned reactor system components and the various cavity structures.

1. Provide arrangement drawings of the reactor vessel support systems in sufficient detail to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural mechanical, and thermal-hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. Consider all postulated breaks in the reactor coolant piping system, including the following locations:

   a. Steam line nozzles to piping terminal ends.
   b. Feedwater nozzle to piping terminal ends.
   c. Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

   Provide an assessment of the effects of asymmetric pressure differentials* on the systems and components listed above in combination with all external loadings including safe shutdown earthquake loads and other

*Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the component and the wall, and transient differential pressures across the core barrel within the reactor vessel.

faulted condition loads for the postulated breaks described above. This assessment may utilize the following mechanistic effects as applicable:

a. Limited displacement -- break areas
b. Fluid-structure interaction
c. Actual time-dependent forcing function
d. Reactor support stiffness
e. Break opening times.

4. If the results of the assessment on item 3 above indicate loads leading to inelastic action of these systems or displacement exceeding previous design limits, provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect of the load transmitted to the backup structures to which these systems are attached.
5. For all analyses performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.
6. Demonstrate that safety-related components will retain their structural integrity when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.
7. Demonstrate the functional capability of any essential piping when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

The applicant has outlined his approach for determining the forcing functions considered in the system and component dynamic analyses of reactor structures for normal operation and anticipated transients. These methods are a combination of analytical methods and predictions based on data from previously tested reactor internals of a similar design. The forcing function information is combined with dynamic modal analysis to form a basis for interpretation of the pre-operational and initial startup test results. Modal stresses are calculated and relationships are obtained between sensor responses and peak component stresses for each of the lower modes.

3.9.3.3-2, page 3.9-106

Provide justification for using a modified static ~~anlaysis~~ analysis on the safety relief valve piping in the suppression pool and explain what is used for the "conservative dynamic load factor" in the analysis.

Provide the time-history transient forces resulting from the SRV actuation used in the SRV piping and support design including the loads developed from the discharging water slug.

Discuss the types of supports used on the SRV piping in both the drywell and suppression pool and provide drawings of the supports.

Provide the type of safety relief valves used in the plant , the valve opening time, and the sequences of valve actuation used in the analysis.

3.9.3.4.6, page 3.9-113

Are the stress due to differential anchor movements considered as primary or secondary stresses for BOP supports?

TO ALL APPLICANTS:

Due to a long history of problems dealing with inoperable and incorrectly installed snubbers, and due to the potential safety significance of failed snubbers in safety related systems and components, it is requested that maintenance records for snubbers be documented as follows:

Pre-service Examination

A pre-service examination should be made on all snubbers listed in tables 3.7-4a and 3.7-4b of Standard Technical Specifications 3/4.7.9 This examination should be made after snubber installation but not more than six months prior to initial system pre-operational testing, and should as a mimimum verify the following:

(1) There are no visible signs of damage or impaired operability as a result of storage, handling, or installation.

(2) The snubber location, orientation, position setting, and configuration (attachments, extensions, etc.) are according to design drawings and specifictions.

(3) Snubbers are not seized, frozen or jammed.

(4) Adequate swing clearance is provided to allow snubber movement.

(5) If applicable, fluid is to the recommended level and is not leaking from the snubber system.

(6) Structural connections such as pins. fasteners and other connecting hardware such as lock nuts, tabs, wire, cotter pins are installed correctly.

If the period between the initial pre-service examination and initial system pre-operational test exceeds six months due to unexpected situations, re-examination of items 1,4, and 5 shall be performed. Snubbers which are installed incorrectly or otherwise fail to meet the above requirements must be repaired or replaced and re-examined in accordance with the above criteria.

Pre-Operational Testing

During pre-operational testing, snubber thermal movements for systems whose operating temperature exceeds 250° F should be verified as follows:

(a) During initial system heatup and cooldown, at specified temperature intervals for any system which attains operating temperature, verify the snubber expected thermal movement.

(b) For those systems which do not attain operating temperature, verify via observation and/or calculation that the snubber will accommodate the projected thermal movement.

(c) Verify the snubber swing clearance at specified heatup and cooldown intervals. Any discrepencies or inconsistencies shall be evaluated for cause and corrected prior to proceeding to the next specified interval.

The above described operability program for snubbers should be included and documented by the pre-service inspection and pre-operational test programs.

The pre-service inspection must be a prerequisite for the pre-operational testing of snubber thermal motion. This test program should be specified in Chapter 14 of the FSAR.

110.0 MECHANICAL ENGINEERING BRANCH

It is the staff's position that all essential safety-related instrumentation lines should be included in the vibration monitoring program during pre-operational or start-up testing. We require that either a visual or instrumented inspection (as appropriate) be conducted to identify any excessive vibration that will result in fatigue failure.

Provide a list of all safety-related small bore piping and instrumentation lines that will be included in the initial test vibration monitoring program.

The essential instrumentation lines to be inspected should include (but are not limited to) the following:

a) Reactor pressure vessel level indicator instrumentation lines (used for monitoring both steam and water levels).

b) Main steam instrumentation lines for monitoring main steam flow (used to actuate main steam isolation valves during high steam flow).

c) Reactor core isolation cooling (RCIC) instrumentation lines on the RCIC steam line outside containment (used to monitor high steam flow and actuate isolation).

d) Control rod drive lines inside containment (not normally pressurized but required for scram).

[redacted] MECHANICAL ENGINEERING BRANCH

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series tc form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems thus causing an inter-system LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCO) are required to be added to the technical specifications which will require corrective action i.e., shutdown or system isolation when the final approved leakage limits are not met. Also surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing interval should average to be approximately one year. Leak testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance and etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting valve would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by case basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

MECHANICAL ENGINEERING BRANCH
DRAFT SAFETY EVALUATION REPORT
PERRY NUCLEAR POWER PLANT UNIT I

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2.1 Seismic Classification

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," of 10 CFR Part 50, Appendix A, in part, requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without loss of capability to perform their safety function. These plant features are those necessary to assure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to 10 CFR Part 100 guideline exposures. The earthquake for which these plant features are designed is defined as the safe shutdown earthquake (SSE) in 10 CFR Part 100, Appendix A. The SSE is based upon an evaluation of the maximum earthquake potential and is that earthquake which produces the maximum vibratory ground motion for which structures, systems, and components important to safety are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated seismic Category I in Regulatory Guide 1.29. Regulatory Guide 1.29, "Seismic Design Classification," is the principal document used in our review for identifying those plant features important to safety which, as a minimum, should be designed to seismic Category I requirements. Our review of the seismic classification of structures, systems, and components (excluding electrical features) of Perry was performed in accordance with the guidance in Standard Review Plan 3.2-1, "Seismic Classification."

The structures, systems, and components important to safety of Perry that are required to be designed to withstand the effects of an SSE and remain functional have been identified in an acceptable manner in Table 3.2-1 of the Final Safety Analysis Report. Table 3.2-1, in part, identifies major components in fluid systems, mechanical systems, and associated structures designated as seismic Category I. In addition, piping and instrumentation diagrams in the Final Safety Analysis Report identify the interconnecting piping and valves and the boundary limits of each system classified as seismic Category I. We have reviewed Table 3.2-1 and the fluid system piping and instrumentation diagrams and have some question concerning part of this table.

It states in the FSAR that structures, components and systems designated as Safety Class 1, 2, or 3 are classified as seismic Category I except for some portions of the radioactive waste treatment handling and disposal systems. There are several items in Table 3.2-1 that conflict with this statement.

"The seismic classification indicated in Table 3.2-1 meets the requirement of Regulatory Guide 1.29." It is also stated in Section 1.8 that the Perry plant complies with all of the requirements of Regulatory Guide 1.29. Does this mean that seismic Category I cooling water is provided to the recirculation pumps during normal operation and o. ;wing a LOCA?

What design requirements were used in the design of the reactor pressure vessel skirt and the core support structures?

Quality assurance requirements should be addressed in Table 3.2-1.

The non-seismic classification of the control rods should be justified. Note 7 does not apply to the control rods.

Provide an explanation for the "I, NA" seismic classification for relief valve discharge piping.

How much of the main steam piping, between the M.O. stop valve and the turbine stop valve, is located in the Auxillary Building?

There appears to be a discrepancy in the seismic classification of the discharge tunnel. The discharge tunnel and the diffusor nozzle are seismic Category I. The tunnel entrance structure and downshaft are not. Provide clarification for this apparent contradiction.

What is the seismic classification of the Containment Vessel Cooling Units?

Note 19 is an exception to Regulatory Guide 1.29 and should be included in Section 1.8.

Based upon our review of FSAR Section 3.2.1 and subject to the satisfactory resolution of the open items, our findings will be as follows.

We have reviewed Table 3.2-1 and the fluid system piping and instrument diagrams and we conclude that the structures, systems, and components important to safety of Perry have been properly classified as seismic Category I items in conformance with Regulatory Guide 1.29, Revision 1.

All other structures, systems, and components that may be required for operation of the facility are not required to be designed to seismic Category I requirements, including those portions of Category I systems such as vent lines, fill lines, drain lines, and test lines on the downstream side of isolation valves and portions of these systems which are not required to perform a safety function.

We conclude that the structures, systems, and components important to safety of Perry that are within the scope of the Mechanical Engineering Branch and are designed to withstand the effects of an SSE and remain functional are properly classified as seismic Category I items in accordance with Regulatory Guide 1.29 and constitutes an acceptable basis for satisfying, in part, the requirements of General Design Criterion 2, and is, therefore, acceptable.

3.3.3 System Quality Group Classification

General Design Criterion 1, "Quality Standards and Records," of 10 CFR Part 50, Appendix A requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. These fluid system, pressure-retaining components are part of the reactor coolant pressure boundary and other fluid systems important to safety, where reliance is placed on these systems: (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintain it in a safe shutdown condition, and (3) to retain radioactive material. Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," is the principal document used in our review for identifying on a functional basis the components of those systems important to safety that are Quality Groups B, C, and D. Section 50.55a of 10 CFR Part 50 identifies those American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Class 1 components that are part of the reactor coolant pressure boundary (RCPB). Conformance of these RCPB components with Section 50.55a of 10 CFR Part 50 is discussed in Section 5.2.1.1 of this Safety Evaluation Report. These RCPB components are designated in Regulatory Guide 1.26 as Quality Group A. Certain other RCPB components which meet the exclusion requirements of footnote 2 of the rule are classified Quality Group B in accordance with Regulatory Guide 1.26. Our review of the quality group classification or pressure-retaining components of fluid systems important to safety for Perry was performed in accordance with the guidance in Standard Review Plan 3.2.2, "System Quality Group Classification."

The systems and components important to safety of Perry have been identified in an acceptable manner in Table 3.2-1 of the Final Safety Analysis Report. Table 3.2-1, in part, identifies the major components in fluid systems such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves and mechanical systems, such as cranes, refueling platforms, and other miscellaneous handling equipment. In addition, the piping and instrumentation diagrams in the Final Safety Analysis Report identify the classification boundaries of the interconnecting piping and valves.

We have reviewed the applicant's use of the NRC Quality Group system in Table 3.2-1 and on the system piping and instrumentation diagrams and we conclude the pressure-retaining components of fluid systems important to safety have been properly classified and meet the guidance in Regulatory Guide 1.26, Revision 2.

We conclude that the applicant's classification of fluid system pressure retaining components important to safety complies with Standard Review Plan Section 3.2.2, Regulatory Guide 1.26 and satisfies the applicable portions of General Design Criterion 1.

Draft Safety Evaluation Report
PERRY NUCLEAR POWER PLANT

Auxiliary Systems Branch

3.4.1 Flood Protection

In order to assure conformance with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" and 10 CFR Part 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Section IV.C., our review of the overall plant flood protection design included all systems and components whose failure due to flooding could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

The applicant has sited the plant (at elevation 620 ft mean sea level (msl)) on a bluff on the shore of Lake Erie approximately 48 ft above the mean lake elevation (572 ft msl) thus providing a "Dry Site" as defined in Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants." In addition, the applicant has provided a seismic Category I pressure relief underdrain system to control the level of the water table at elevation 590 ft msl without pumping. Calculations show that the maximum surge flood (setup plus wave runup) elevation of Lake Erie to be 608 ft msl. Calculations of flooding due to a probable maximum precipitation (PMP) show that the water level on plant grade does not exceed the floor level of the buildings. Flooding from streams or rivers is not possible because of the nature of the plant site. (Refer to Section 2.4 of this SER for further discusson on flooding.) Thus the guidelines of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," are met.

Safety-related systems and components that must be protected against flooding have been identified and are located in safety-related structures. All penetrations in these structures below the 590-ft level are watertight. These structures are also provided with waterstops in all construction joints below level 590 ft and are provided with waterproof coatings. Within these structures, protection against flooding from failures in fluid piping systems as identified in the guidelines of Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," is provided by placing critical equipment in watertight cubicles. This feature is discussed in more detail under Section 9.3.3 of this SER.

Based on our review of the design criteria and bases, and safety classification of safety-related systems, structures, and components necessary for a safe plant shutdown during and following flood conditions, we conclude that the design of the facility for flood protection conforms to the requirements of General Design Criterion 2 and 10 CFR Part 100, Appendix A with respect to protection against natural phenomena and the guidelines of Regulatory Guides 1.59 and 1.102 concerning design basis floods and flood protection and is, therefore, acceptable, pending review of the interval flood analysis.

[The information promised in reply to question 410.5 concerning internal flood analysis was received; however, we have not completed our review of this item.] We will report on the resolution of this matter in a supplement to this SER.

3.5.1.1 Internally Generated Missiles (Outside Containment)

General Design Criterion 4, "Environmental and Missile Design Bases," requires protection of plant structures, systems, and components outside containment, whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown, against postulated missiles associated with plant operation. The missiles considered in this evaluation include those missiles generated by rotating or pressurized (high-energy fluid system) equipment. The protection is provided by any one or a combination of compartmentalization, barriers, separation, and equipment design. The primary means of providing protection to safety-related equipment from damage resulting from internally generated missiles is through the use of plant physical arrangement. Safety-related systems are physically separated from nonsafety-related systems and redundant components of safety-related systems are physically separated such that potential missiles could not damage both trains of safety-related equipment. Stored spent fuel in the intermediate building is protected from damage by internal missiles which could result in radioactive release in accordance with the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," by the fuel pool walls and by preventing the location of high energy piping system or rotating machinery in the vicinity of new or spent fuel.

The applicant provided an evaluation of potential missile sources on the basis that a single failure could result in their becoming potential missiles. The potential missiles resulting from this analysis are instrumentation wells and resistance temperature detectors in high energy systems. Maximum velocities, weights and postulated trajectories for these missiles were determined. The analysis verified that plant features, walls and redundant system separation prevented these missiles from causing adverse effects on safety-related systems and components. We concur with the applicant's assumptions and evaluation for potential missiles outside containment.

Protection of safety-related equipment and stored fuel from the effects of turbine missiles including compliance with the guidelines of Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," is discussed in Section 3.5.1.3 of this SER.

We have reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown and prevent unacceptable radiological release in the event of internally generated missiles outside containment. Based on the above, we conclude that the design is in conformance with the requirements of General Design Criterion 4 as it relates to protection against internally generated missiles, and the guidelines of Regulatory Guide 1.13 as it relates to protection of spent fuel from internal missiles and is, therefore, acceptable.

3.5.1.2 Internally Generated Missiles (Inside Containment)

All plant structures, systems, and components (SSC) inside containment whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown must be protected against the effects of internally generated missiles in accordance with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases." Potential missiles that could be generated inside containment are from failures of rotating components, pressurized component (high-energy fluid system) failures, and gravitational effects.

The applicant's analysis of rotating equipment (pump impellers, compressors, fan blades, motors and couplings) failures indicates that equipment design prevents such components from becoming sources of potential missiles. Generation of missiles from overspeed of both the motor and impeller of the reactor recirculation pump following a postulated full double-ended pipe break in either the suction or discharge line of the pump is a generic problem which is being reviewed by the staff under Task Action Plan B-68, "Pump Overspeed During a LOCA." At this time, we believe that the probability of such an event that would result in damage to safety-related equipment is acceptably low for licensing and operation of this plant. Should the results of our generic study indicate the need for any design modifications, the applicant will be required to satisfy these requirements.

The applicant considered the following for potential missiles from pressurized high-energy fluid systems: valve bonnets; safety relief valves; unrestrained piping such as instrument connections, vents and drains; valve stems; temperature detectors (thermowells); nuts and bolts; and pressurized gas bottles and accumulators. The applicant performed analyses to demonstrate that the design of the above components either prevents the generation of missiles as a result of a single failure, or, if generated, the missiles either have insufficient energy to cause unacceptable damage, or else adequate compartmentalization, separation or barriers are provided for protection of safety-related equipment. Missile characteristics, trajectory and impact area were included in this analysis as applicable. The applicant also analyzed the effects of secondary missiles generated by those primary missiles determined above. This analysis indicated that secondary missiles will no affect safety-related systems or components. We have reviewed the results of the applicant's internally generated missile analysis inside containment and agree with the conclusion that unacceptable damage to safety-related equipment will not occur.

In addition, the applicant evaluated the potential for gravitational missiles inside containment. All nonsafety-related components are supported to prevent their collapse in an SSE.

Spent fuel stored within the containment is located in an area separated from the potential internal missile sources previously identified. The upper containment pool walls also provide protection from potential internally generated missiles. Thus the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," are satisfied.

We have reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown and prevent unacceptable radiological release in the event of internally generated missiles inside containment. Based on the above, we conclude that the design is in conformance with General Design Criterion 4 as it relates to protection against internally generated missiles, and Regulatory Guide 1.13 as it relates to protection of spent fuel from internal missiles and is, therefore, acceptable.

3.5.1.4 Missiles Generated by Natural Phenomena

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components essential to safety be designed to withstand the effects of natural phenomena, and General Design Criterion 4, "Environmental and Missile Design Bases," requires that these same plant features be protected against missiles. The missiles generated by natural phenomena that are of concern are those resulting from tornadoes. The applicant has identified a spectrum of missiles for a tornado region 1 site as identified in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The spectrum includes the weight, velocity, kinetic energy, impact area and height, penetration depth, and minimum available concrete thickness providing protection. We have evaluated this spectrum and conclude that it is representative of missiles at the site and is, therefore, acceptable. A discussion of the protection afforded safety-related equipment from the identified tornado missiles including compliance with the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," is provided in Section 3.5.2 of this SER. A discussion of the adequacy of barriers and structures designed to withstand the effects of the identified tornado missiles is provided in Section 3.5.3 of this SER. Based upon our review of the tornado missile spectrum, we conclude that the spectrum was properly selected and meets the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and missiles and the guidelines of Regulatory Guides 1.76 and 1.117 with respect to identification of missiles generated by natural phenomena and is, therefore, acceptable.

3.5.2 Structures, Systems, and Components to be Protected from Externally Generated Missiles

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that all structures, systems, and components essential to the safety of the plant be protected from the effects of natural phenomena, and General Design Criterion 4, "Environmental and Missile Design Bases," requires that all structures, systems, and components essential to the safety of the plant be protected from the effects of externally generated missiles. The Perry site is located in tornado region 1 as identified in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The tornado missile spectrum is discussed in Section 3.5.1.4 of this SER.

The applicant has identified all safety-related structures, systems, and components requiring protection from externally generated missiles. All safety-related structures are designed to withstand postulated tornado generated missiles without damage to safety-related equipment [except for ventilation openings described in Section 9.4.3 of this SER]. All safety-related systems and components and stored fuel are located within tornado-missile-protected structures or are provided with tornado missile barriers. Buried safety-related systems such as piping and electrical circuits are adequately protected by the overlaying earth and adequate manholes where necessary. The ultimate heat sink, Lake Erie, has inherent protection against natural phenomena. Pending resolution of the concern identified in Section 9.4.3 of this SER, the requirements of General Design Criteria 2 and 4 with respect to missile protection and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and 1.117, "Tornado Design Classification," concerning tornado missile protection for safety-related structures, systems, and components including stored fuel and the ultimate heat sink are met. Protection from low-trajectory turbine missiles including compliance with Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," is discussed in Section 3.5.1.3 of this SER.

Based on the above and pending resolution of the concern identified in Section 9.4.3 of this SER, we conclude that the applicant's list of safety-related structures, systems, and components to be protected from externally generated missiles and the provisions in the plant design providing this protection are in accordance with the requirements of General Design Criteria 2 and 4 with respect to missile and environmental effects and the guidelines of Regulatory Guides 1.13, 1.27, 1.115 and 1.117 concerning protection of safety-related plant features including stored fuel and the ultimate heat sink from tornado missiles and is, therefore, acceptable.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

The review performed under this section pertains to the applicant's program for protecting safety-related components and structures against the effects of postulated pipe breaks both inside and outside containment. The effect that breaks or cracks in high or moderate energy fluid systems would have on adjacent safety-related components or structures has been analyzed with respect to jet impingement, pipe whip, and environmental effects. Several means are used to assure the protection of these safety-related items. They include physical separation, enclosure within suitably design structures, the use of pipe whip restraints, and the use of equipment shields.

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

The staff's guidelines for meeting the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," concerning protection against postulated piping failures in high-energy and moderate-energy fluid systems outside containment are contained in Branch Technical Position ASB 3-1, "Protection Against Postulated Failures in Fluid Systems Outside Containment." The applicant has identified all high- and moderate-energy piping systems in accordance with these guidelines and has also identified those systems requiring protection from postulated piping failures. The plant design accommodates the effects of postulated pipe breaks in high-energy fluid piping systems outside containment with respect to pipe whip, jet impingement and resulting reactive forces, and environmental effects, and the effects of postulated cracks in moderate-energy fluid systems outside containment with respect to jet impingement, flooding and other environmental effects. The [illegible] used to protect safety-related systems and components throughout the plant include physical separation, enclosure in suitably designed structures or compartments, drainage systems, pipe whip restraints, equipment shields, and equipment environmental qualification as required.

The applicant analyzed high-energy piping systems for the effects of pipe whip, jet impingement, and environmental effects on safety-related systems and structures. For moderate-energy systems, protection of safety-related systems from the jet, flooding and other environmental effects due to critical cracks is incorporated into the plant design. We have reviewed the applicant's analysis and we conclude that the protection provided against pipe failure outside containment is in conformance with the guidelines of Branch Technical Position ASB 3-1.

The main steam lines which include the outboard isolation valves and the feedwater lines are all located in the common auxiliary building steam tunnel and have been classified as part of the break exclusion boundary. The applicant has performed a subcompartment analysis for the steam tunnel and the main steam lines in order to assure that the resulting jet impingement and environmental effects from a postulated full circumferential pipe break in one of these lines or a feedwater line will not result in adverse consequences. The results of this analysis indicate that the steam tunnel structural integrity is not affected by the pressure increase from the resulting blowdown. Main steam isolation valve (MSIV) functional capability is assured by the environmental qualification of system components to the expected condition of 330°F for one hour. The analysis also indicated that the MSIV closure will terminate the blowdown at 5.5 seconds.

The plant has the ability to sustain a high-energy pipe break coincident with a single active failure in essential systems and retain the capability for safe cold shutdown. For postulated pipe failures, the resulting environmental effects do not preclude the habitability of the control room, the accessibility of other areas that have to be manned during and following an accident, and the loss of function of electric power supplies and controls and instrumentation needed to complete a safety action. Further discussion of the environmental qualification of safety-related equipment is contained in Section 3.11 of this SER.

Based on our review, we find that the applicant has adequately designed and protected areas and systems required for safe plant shutdown following postulated events, including the combination of pipe failure and single active failure. We conclude that the plant design meets the requirements of General Design Criterion 4 and the criteria set forth in Branch Technical Position ASB 3-1 with regard to the protection of safety-related systems and components from a postulated high-energy line break and with regard to the protection of safety-related systems and components from a postulated moderate-energy line failure.

We further conclude that the plant design for the protection of safety-related equipment against dynamic effects associated with the postulated rupture of piping outside containment is acceptable.

3.6.2 Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping

Our review under Standard Review Plan 3.6.2 was concerned with the locations chosen by the applicant for postulating piping failures. We also reviewed the size and orientation of these postulated failures and how the applicant calculated the resultant pipe whip and jet impingement loads which might affect nearby safety related components.

Standard Review Plan 3.6.2 also sets forth certain criteria for the analysis and subsequent in-service inspection of high energy piping in the break exclusion area of containment penetration. Breaks need not be postulated in those portions of piping that meet the requirements of the ASME Code, Section III, Subarticle NE-1120 and the additional design requirements outlined in Branch Technical Position MEB 3-1. Additional in-service inspection is also required for those portions of piping.

The following discusses open issues found in our review of FSAR Section 3.6.2. It concludes with our findings contingent upon resolution of all open issues.

continued on next page.

In Section 3.6.1 references are made to "elastic/plastic pipe whip restraints or pipe supports which eliminate pipe whip damage." Details of how pipe supports are designed for pipe whip protection and an example of such an analysis are needed.

Pipe whip need only be considered in those high energy piping systems having sufficient capacity to develop a jet stream. The means for determining high and moderate energy lines is found in Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment". This criteria has been used correctly by the applicant. Some additional information is required to clarify this section. How is it determined that the "internal energy level associated with whipping is insufficient to impair the safety function of any system or component to an unacceptable level"? Details should be provided of any flow restrictors used. Methods used to determine fluid reservoirs with sufficient capacity to develop a jet stream should also be provided.

~~Pipe breaks are to be postulated at the terminal ends of piping. The definition of terminal ends should be expanded to show that rotational movements from dynamic or static loadings are also restrained.~~

For determining stresses or fatigue usage factors that require a pipe break to be postulated, plant loadings are to be those resulting from normal and upset conditions plus an OBE. Assurances must be provided that loads due to SRV actuation and discharge are included in the upset conditions.

For ASME, Section III, Class 1 piping designed to seismic Category I standards, breaks due to stress are to be postulated at the following locations:

(1) If Eq. (10), as calculated by Paragraph NB-3653, ASME Code Section III, exceeds 2.4 S_m, then Eqs. (12) and (13 must be evaluated. If either Eq. (12) or (13) exceeds 2.4 S_m, a break must be postulated. In other words, a break is postulated if

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} > 2.4\ S_m$$

or

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (13)} > 2.4\ S_m$$

(2) Breaks must also be postulated at any location where the cumulative usage factor exceeds 0.1.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

Any deviations from the above criteria must be justified.

Are there any high energy Class 2, Class 3 or B31.1 lines? If so, what criteria is used for postulating breaks in these lines?

Any instances with limited break openings or break opening times exceeding one millisecond must be identified. Any analytical methods, representing test results or based on a mechanistic approach, used to justify the above must be provided and explained in detail. This applies to containment and annulus pressurization as well as general pipe break.

For those portions of ASME, Section III, Class 1 piping designed to seismic Category I standards and included in the break exclusion area breaks need not be postulated providing the following stress criteria are met.

(1) If Eq. (10) as calculated by Paragraph NB-3653, ASME Code, Section III does not exceed 2.4 S_m, a break need not be postulated.

(2) If Eq. (10) does exceed 2.4 S_m, then Eqs. (12) and (13) must be evaluated. If neither Eq. (12) or (13) exceeds 2.4 S_m, a break need not be postulated. In other words, a break need not be postulated if:

$$\text{Eq. (10)} < 2.4\ S_m$$

or

$$\text{Eq. (10)} > 2.4\ S_m \text{ and Eq. (12)} < 2.4\ S_m$$

and

$$\text{Eq. (13)} < 2.4\ S_m$$

(3) Breaks need not be postulated as long as the cumulative fatigue usage factor is less than 0.1.

(4) For plants with isolation valves inside containment, the maximum stress, as calculated by Eq. (9) in ASME Code Section III, Paragraph NB-3652 under the loadings of internal pressure, deadweight and a postulated piping failure of fluid systems upstream or downstream of the containment penetration area must not exceed 2.25 S_m.

The above criteria is evaluated under loadings resulting from normal and upset plant conditions including the OBE.

In addition, augmented inservice inspection is required on all piping in the break exclusion area.

The applicant must provide assurances that their criteria for piping in the break exclusion areas complies with the requirements outlined above and those of Standard Review Plan 3.6.2.

Are there any Class 2, Class 3, or B31.1 piping in the break exclusion areas. If so, what criteria is used in their design?

A list of all systems included in the break exclusion areas must be included in the FSAR. In addition, break exclusion areas should be shown on the appropriate piping drawings.

Provide an example of the detailed stress analysis done on a welded attachment to a process pipe. In addition, provide details of the stress analysis done for the head fitting for the main steam line.

When providing protection from pipe whip, assurances must be provided that all potential targets are examined. Provide a definition for limits of strain which are similar to strain levels allowed in restraint plastic members.

"Piping systems are designed so that plastic instability does not occur in the pipe at the design dynamic and static loads unless damage studies are performed which show the consequences do not result in direct damage to any essential system or component." Provide a list of locations where this technique has been used and an example of the studies performed.

~~The FSAR states that details of the structural design and analysis of the steam tunnel are provided in Section 3.6.2.3. Provide the details of this analysis as indicated in the FSAR.~~

When evaluating the effects of jet impingement loads it is the staff's position that all potential targets must be evaluated. Assurances must be provided that your analysis has considered all potential targets. What service limits are used for piping when evaluating jet impingement loads?

Reference is made to the use of a suitable dynamic load factor (DLF). Provide an example of its use. How is it determined that it is suitable?

In the discussion about snubbers, reference is made to "other simultaneous loads". It further states that these loads are combined by SRSS. What are these loads?

"Piping integrity usually does not depend upon the pipe whip restraints for any loading combination." List all those locations and loading combinations where it does. What service limits are used in the design of the pipe whip restraints?

During hot functional testing what critical locations inside containment are monitored?

Standard Review Plan 3.6.2 allows a 10% increase in yield strength to account for strain rate effects. Any locations where an increase in the yield or ultimate strength greater than 10% has been used must be identified. Justification for any increase greater than 10% must also be provided.

Our review of Section 3.6.2 includes all tables and figures. We have several questions pertainint to tables and figures.

Provide a schedule for the completion of any table that is incomplete. Are all postulated break locations in the recirculation system shown (Figure 3.6-66)? Where are breaks postulated in these figures (Figures 3.6-71, 3.6-73, 7.3-74, 3.6-77, 3.6-78, 3.6-79, 3.6-80)? Indicate the location of valves in this line (Figure 3.6-75).

Based on our review of FSAR Section 3.6.2 and subject to the satisfactory resolution of the identified open items, our findings will be as follows:

The applicant has proposed criteria for determining the location, type, and effects of postulated pipe breaks in high energy piping systems and postulated pipe cracks in moderate energy piping systems. The applicant has used the effects resulting from these postulated pipe failures to evaluate the design of systems, components, and structures necessary to safely shut the plant down and to mitigate the effects of these postulated piping failures. The applicant has stated that pipe whip restraints, jet impingement barriers, and other such devices will be used to mitigate the effects of these postulated piping failures.

We have reviewed these criteria and have concluded that they provide for a spectrum of postulated pipe breaks and pipe cracks which includes the most likely locations for piping failures, and that the types of breaks and their effects are conservatively assumed. We find that the methods used to design the pipe whip restraints provide adequate assurance that they will function properly in the event of a postulated piping failure. We further conclude that the use of the applicant's proposed pipe failure criteria in designing the systems, components, and structures necessary to safely shut the plant down and to mitigate the consequences of these postulated piping failures provides reasonable assurance of their ability to perform their safety function following a failure in high or moderate energy piping systems. The applicant's criteria comply with Standard Review Plan Section 3.6.2 and satisfy the applicable portions of General Design Criterion 4.

3.7.3 Seismic Subsystem Analysis

The review performed under Standard Review Plan Section 3.7.3 included the applicant's dynamic analysis of all seismic Category I piping systems. In addition to operating transient loads such as suppression pool loads, this analysis also considers abnormal loadings such as an earthquake.

For the dynamic analysis of seismic Category I piping, each pipe line was idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system was determined using the elastic properties of the pipe. This includes the effects of torsional, bending, shear, and axial deformations as well as change in stiffness due to curved members. Next, the mode shapes and the undamped natural frequencies were obtained. The dynamic response of the system was calculated by using the response spectrum method of analysis. For a piping system which was supported at points with different dynamic excitations, the response spectrum analysis was performed using the envelope response spectrum of all support points. Alternately, the multiple support excitation analyses methods may have been used where separate acceleration time-histories were applied to each piping system support points.

The following discusses open issues found in our review of FSAR Section 3.7.3. It concludes with our findings which are contingent upon the resolution of all open issues.

The discussion on "Different Seismic Movement of Interconnected Components" requires some clarification. "The stresses thus obtained for each natural mode are then superimposed for all modal displacements of the structure by the SRSS (square root sum of the squares) method." Provide an example of this type of analysis.

What criteria is used to determine whether or not a mode is significant?

"When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacements as obtained above are treated as secondary stresses." Does this statement pertain to piping or supports?

"Seismic analyses were performed for those subsystems that could be modeled to correctly predict the seismic response." What procedure was used for the other systems? Provide an example of those systems and the analysis done.

It is the staff's position that closely spaced modes be combined by one of the procedures identified in Regulatory Guide 1.92. What procedure is used in the BOP design to account for closely spaced modes? What is meant by "Closely spaced phase modes"? Show how modal phasing can be determined from a response spectrum analysis.

Standard Review Plan 3.7.3 requires that 5 OBEs of 10 cycles each be used for design. Any deviations from the requirements of the SRP must be justified. How many OBE cycles are considered in the NSSS and BOP designs?

~~"If the torsional effect is expected to cause pipe stresses less than 500 psi, this effect may be neglected." How are the torsional stresses estimated? Justify the use of 500 psi as a limit.~~

In the discussion concerning the modeling of piping part (a) discussing decoupling of the main steam and branch lines is not a criteria.

Mention is made of using 33 hertz as a cutoff frequency for seismic analysis. At some point in the FSAR the applicant must address the frequencies of 50 to 60 hertz and greater that come from the suppression pool hydrodynamics.

"For flexible equipment, the equivalent static load is taken as the product of 1.5 times the equipment mass and the peak floor response spectrum value." Regulatory Guide 1.100 allows the use of the 1.5 factor for verifying the integrity of frame type structures. For equipment having configurations other than a frame type structure, justification is required for the use of the 1.5 factor.

When using the double sum method to combine modal responses, the product of the responses of the closely spaced modes should be taken as an absolute value.

Assurances must be provided that the modeling of valves with offset motor operators is detailed enough to provide acceleration values to be used for valve qualification.

"In addition, the effects of modes not included are added to the SRSS response as one term, using the acceleration at the highest frequency from the SRSS response under 33 hertz to obtain the total response." Provide an example of the analysis done here.

The information presented in Table 3.7.11 is not straightforward. Provide an explanation of this table.

Based on our review of FSAR Section 3.7.3 and subject to the satisfactory resolution of the identified open items, our findings will be as follows:

The scope of review of the seismic system and subsystem analysis for the Perry plant included the seismic analysis methods for all Category I systems and components. It included review of procedures used for modeling and evaluating Category I systems and components. The review included design criteria and procedures for evaluation of the interaction of non-Category I piping with Category I piping. The review also included criteria and seismic analysis procedures for reactor internals and Category I piping outside containment.

The system and subsystem analyses are performed by the applicant on an elastic basis. Modal response spectrum multidegree of freedom and time history methods form the bases for the analyses of all major Category I systems and components. When the modal response spectrum method is used, governing response parameters are combined by the square root of the sum of the squares rule. However, the absolute sum of the modal responses are used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum codirectional responses is used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. A vertical seismic system dynamic analysis is employed for all systems and components.

We conclude that the seismic system and subsystem analysis procedures and criteria proposed by the applicant provide an acceptable basis for the seismic design of systems and components.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

The review performed under Standard Review Plan Sections 3.9.1 through 3.9.6 pertains to the structural integrity and operability of various safety-related mechanical components in the plant. Our review is not limited to ASME Code components and supports, but is extended ot other components such as control rod drive mechanisms, certain reactor internals, ventilation ducting, cable trays, and any safety-related piping designed to industry standards other than the ASME Code. We review such issues as load combinations, allowable stresses, methods of analysis, summary results, and pre-operational testing. Our review must arrive at the conclusion that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The review performed under Standard Review Plan Section 3.9.1 pertains to the design transients, computer programs, experimental stress analyses and elastic-plastic analysis methods that were used in the analysis of seismic Category I ASME Code and non-Code items.

Additionally, we have contracted with Pacific Northwest Laboratories to perform an independent analysis of a sample piping system in the Perry plant. This analysis will verify that the sample piping system meets the applicable ASME Code requirements. We will report the results of this independent piping analysis in a supplement to this Safety Evaluation Report.

Computer programs were used in the analysis of specific components. A list of the computer programs used in the dynamic and static analyses to determine the structural and functional integrity of these components must be included in the FSAR along with a brief description of each program. Design control measures, which are required by 10 CFR Part 50, Appendix B, require that verification of the computer programs also be included. The applicant has not provided verification for all of the listed computer programs.

In addition, the programs DYREC and DYNAL are not included in the list of computer programs used.

Any reference to the ASME Boiler and Pressure Vessel Code should include the part being referenced.

How many SRV cycles have been used in the design of components and systems for the NSSS and BOP scope? How many ADS cycles?

It is stated that elastic-plastic methods of analysis may be used for some components. We would like to review the analysis procedures that would be used if an elastic-plastic analysis was done.

Based upon our review of FSAR Section 3.9.1 and contingent on the satisfactory resolution of the open items, our findings will be as follows.

The methods of analysis that the applicant has employed in the design of all seismic Category I ASME Code Class 1, 2 and 3 components, component supports, reactor internals, and other non-Code items are in conformance with Standard Review Plan 3.9.1 and satisfy the applicable portions of General Design Criteria 2, 4, 14 and 15.

The criteria used in defining the applicable transients and the computer codes and analytical methods used in the analyses provide assurance that the calculations of stresses, strains, and displacements for the above noted items conform with the current state-of-the-art and are adequate for the design of these items.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The review performed under Standard Review Plan Section 3.9.2 pertains to the criteria, testing procedures, and dynamic analyses employed by the applicant to assure the structural integrity and operability of piping systems, mechanical equipment, reactor internals and their supports under vibratory loadings. Seismic qualification of safety-related mechanical equipment will be reviewed by the Equipment Qualification Branch.

Piping vibration, thermal expansion, and dynamic effects testing will be conducted during a preoperational testing program. The purpose of these tests is to assure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent. During the Perry plant's preoperational and startup testing program, the applicant will test various piping systems for abnormal steady-state or transient vibration and for restraint of thermal growth. This test program must comply with the ASME Code, Section III, paragraphs NB-3622, NC-3622, and ND-3622 which require that the designer be responsible by observation during startup or initial operation, for ensuring that the vibration of piping systems is within the acceptable levels. In addition, pipe whip restraint initial clearances will be checked, as will snubber response. The test program should consist of a mixture of instrumented measurements and visual observation by qualified personnel. The applicant will be required to provide a summary of the results of this test program upon its completion.

The applicant's discussion of the testing program in the FSAR is too general and should be redone. More detail of what will actually be done must be provided. The applicant has not given a clear description of the NSSS acceptance criteria for steady-state piping vibrations. The BOP program has not been adequately described. What are the acceptance criteria for steady-state vibrations? For transient vibration? Will snubbers be checked? To what transients will the piping be subjected? Which lines, if any, will be instrumented? If not instrumented, how will the visual observations be performed and on what size pipe lines? The staff's position is that acceptance limits for vibration should be based on half the endurance limit as defined by the ASME Code at 10^6 cycles.

In the discussion on thermal expansion testing of the main steam line, reference is made to the piping system shaking down after a few thermal expansion cycles. Provide an explanation of this statement.

It is stated in the FSAR that Perry will be the prototype for the 238 BWR/6. Provide a commitment that the testing program will be equivalent to that required by Regulatory Guide 1.20 for prototype reactors.

"In addition to the above components, vibration measurements of the core spray sparger will be measured during the preoperational testing of that system at the designated prototype 251 BWR/6 plant (Grand Gulf)." Show how this will be applicable to Perry.

It appears that some results from Grand Gulf will be used in the evaluation and qualification of reactor internals at Perry. Show that the similarity between the two sets of internals is sufficient to allow direct comparisons.

"These periods will be determined from a comprehensive dynamic model of the RPV and internals with 12 degrees of freedom." It is not clear what is actually done here. How can a model be comprehensive and have only 12 degrees of freedom?

Previous analyses for other nuclear plants have shown that certain reactor system components and their supports may be subjected to previously under-estimated asymmetric loads under the conditions that result from the postulation of ruptures of the reactor coolant piping at various locations.

The applicant has described the design of the reactor internals for blowdown loads only. The applicant should also provide information on asymmetric loads. It is, therefore, necessary to reassess the capability of these reactor system components to assure that the calculated dynamic asymmetric loads resulting from these postulated pipe ruptures will be within the bounds necessary to provide high assurance that the reactor can be brought safely to a cold shutdown condition. The reactor system components that require reassessment shall include:

a. Reactor pressure vessel
b. Core supports and other reactor internals
c. Control rod drives
d. ECCS piping that is attached to the primary coolant piping
e. Primary coolant piping
f. Reactor vessel supports

The following information should be included in the FSAR about the effects of postulated asymmetric LOCA loads on the above mentioned reactor system components and the various cavity structures.

1. Provide arrangement drawings of the reactor vessel support systems in sufficient detail to show the geometry of all principal elements and materials of construction.

2. If a plant-specific analysis will not be submitted for your plant, provide supporting information to demonstrate that the generic plant analysis under consideration adequately bounds the postulated accidents at your facility. Include a comparison of the geometric, structural mechanical, and thermal-hydraulic similarities between your facility and the case analyzed. Discuss the effects of any differences.

3. Consider all postulated breaks in the reactor coolant piping system, including the following locations:

   a. Steam line nozzles to piping terminal ends.

   b. Feedwater nozzle to piping terminal ends.

   c. Recirculation inlet and outlet nozzles to recirculation piping terminal ends.

   Provide an assessment of the effects of asymmetric pressure differentials* on the systems and components listed above in combination with all external loadings including safe shutdown earthquake loads and other faulted condition loads for the postulated breaks described above. This assessment may utilize the following mechanistic effects as applicable:

   a. Limited displacement -- break areas

   b. Fluid-structure interaction

   c. Actual time-dependent forcing function

   d. Reactor support stiffness

   e. Break opening times.

*Blowdown jet forces at the location of the rupture (reaction forces), transient differential pressures in the annular region between the component and the wall, and transient differential pressures across the core barrel within the reactor vessel.

4. If the results of the assessment on item 3 above indicate loads leading to inelastic action of these systems or displacement exceeding previous design limits, provide an evaluation of the inelastic behavior (including strain hardening) of the material used in the system design and the effect of the load transmitted to the backup structures to which these systems are attached.

5. For all analyses performed, include the method of analysis, the structural and hydraulic computer codes employed, drawings of the models employed and comparisons of the calculated to allowable stresses and strains or deflections with a basis for the allowable values.

6. Demonstrate that safety-related components will retain their structural integrity when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

7. Demonstrate the functional capability of any essential piping when subjected to the combined loads resulting from the loss-of-coolant accident and the safe shutdown earthquake.

The applicant has outlined his approach for determining the forcing functions considered in the system and component dynamic analyses of reactor structures for normal operation and anticipated transients. These methods are a combination of analytical methods and predictions based on data from previously tested reactor internals of a similar design. The forcing function information is combined with dynamic modal analysis to form a basis for interpretation of the pre-operational and initial startup test results. Modal stresses are calculated and relationships are obtained between sensor responses and peak component stresses for each of the lower modes.

The applicant has committed to vibrational measurement and inspection programs to be conducted during preoperational and initial startup testing. The applicant should confirm that testing ~~Testing~~ will be in accordance with the guidelines of Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing" for prototype plants.

These tests will be conducted in three phases. These are preoperational tests prior to fuel loading, zero-power tests with fuel, and initial start-up tests. During preoperational testing, steady-state test conditions will include balanced (two-pump) and unbalanced (one-pump) operation of the recirculation system with flow over the full range up to rated flow. Transient flow conditions will include single and dual pump trips from rated flow. Test duration will ensure that a minimum of 10^6 cycles of vibration will be experienced by the critical components. Inspection of internals will be conducted before and after the test. The zero-power tests with fuel are to verify the anticipated effects of the fuel on the vibration response of internals prior to criticality. Test flow conditions will be similar to the preoperational tests. During the initial startup tests, flow conditions will be similar to the other tests except that power will be up to 100 percent of rated. The primary purpose of these tests is to verify the anticipated effect of two-phase flow.

Vibration sensor types will include strain gages, displacement sensors (linear variable transformers), and accelerometers. Accelerometers will be provided with double integration signal conditioning to give a displacement output. Sensor locations and measured parameters will include the following:

Top of shroud head, lateral acceleration and displacement.
Top of shroud, lateral displacement.
Jet pump riser braces, bending and extension strains.
Jet pump diffuser, lateral motion or bending strain.
Control rod drive housings, bending strain.
Incore housings, bending strain.
Core spray internal piping, bending strain.

The applicant will be required to provide a brief summary of the results of this test program upon its completion.

Recently, cracking has been observed in BWR jet pump hold down beams. The resolution of this problem may affect the design or testing of the Perry jet pumps. (See IE Bulletin 80-07.)

Based upon our review of FSAR Section 3.9.2.1 and contingent upon the satisfactory resolution of the open items, our findings will be as follows:

The vibration, thermal expansion, and dynamic effects test program which will be conducted during startup and initial operation on specified high and moderate energy piping, and all associated systems, restraints and supports is an acceptable program. The tests provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design basis flow conditions. In addition, the tests provide assurance that adequate clearnaces and free movement of snubbers exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. The planned tests will develop loads similar to those experienced during reactor operation. This test program complies with Standard Review Plan Section 3.9.2 and constitutes an acceptable basis for fulfilling the applicable requirements of General Design Criteria 14 and 15.

Based upon our review of FSAR Section 3.9.2.3, 3.9.2.4, and 3.9.2.6 and subject to resolution of the above open issue, our findings are as follows:

The preoperational vibration program planned for the reactor internals provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provide adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The integrity of the reactor internals in service is essential to assure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown. The conduct of the preoperational vibration tests is in conformance with the provisions of Regulatory Guide 1.20 and Standard Review Plan Section 3.9.2, and satisfies the applicable requirements of General Design Criteria 1 and 4.

The applicant has analyzed the reactor, its internals, and unbroken loops of the reactor coolant pressure boundary, including the supports, for the combined loads due to a simultaneous loss-of-coolant accident and safe shutdown earthquake. We cannot finalize our review in this area until the applicant submits the information requested under the new loads program. (annulus pressurization)

Based upon our review of the FSAR Section 3.9.2.5 and subject to resolution of any open items, our findings are as follows:

The dynamic system analysis performed by the applicant provides an acceptable basis for confirming the structural design adequacy of the reactor, its internals, and unbroken piping loops to withstand the combined dynamic loads of postulated loss of coolant accident (LOCA) and the safe shutdown earthquake (SSE). The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals do not exceed the allowable stress and strain limits for the materials of construction, and that the resulting deflections or displacements at any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found to be compatible with those used for the system analysis. The proposed combinations of component and system analyses are, therefore, acceptable.

The assurance of structural integrity under LOCA and SSE conditions for the most adverse postulated loading event provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events. Accomplishment of the dynamic system analysis constitutes an acceptable basis for complying with Standard Review Plan Section 3.9.2 and for satisfying the applicable requirements of General Design Criteria 2 and 4.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

Our review under Standard Review Plan Section 3.9.3 is concerned with the structural integrity and operability of pressure-retaining components, their supports, and core support structures which are designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, or earlier industrial standards. This review is divided into three parts, each of which is discussed briefly below.

The first area of review is the subject of load combinations methodology used in load/response combinations and allowable stresses. The applicant has provided a commitment that all ASME Class 1, 2, and 3 components, component supports, core support structures, control rod drive components, and other reactor internals have been analyzed or qualified in accordance with the referenced loading combinations.

Several references are made throughout this section to allowable stresses for bolting. Specifically, what allowable stress limits are used for bolting for (a) equipment anchorage, (b) component supports, and (c) flanged connections? Where are these limits defined?

Are there any Class 1 systems in the BOP scope of responsibility?

The tables in this section provide the major source of information. These tables should be carefully examined by the applicant to ensure clarity and continuity.

Based on our review of FSAR 3.9.3.1 and contingent upon the satisfactory resolution of the open issue, our findings will be 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 potulated 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.

The second area of review in this section concerns the criteria used by the applicant in designing its ASME Class 1, 2, and 3 safety and relief valves, their attached piping, and their supports. We have specifically reviewed the applicant's compliance with Regulatory Guide 1.67, "Installation of Overpressure Protective Devices". We require further clarification of the analyses performed on the SRV piping and supports.

Based upon our review of FSAR section 3.9.3.3 and contingent upon the satisfactory resolution of the open items, our findings will be as follows:

The criteria used in the design and installation of ASME Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function. The criteria used for the design and installation of ASME Class 1, 2, and 3 overpressure relief devices constitute an acceptable basis for meeting the applicable requirements of General Design Criteria 1, 2, 4, 14, and 15 and are consistent with those specified in Regulatory Guide 1.67 and Standard Review Plan Section 3.9.3.

The third area of our review in this section was the criteria used by the applicant in the design of ASME Class 1, 2, and 3 component supports. All component supports have been designed in accordance with Subsection NF of the ASME Code, Section III.

We have reviewed the applicant's design criteria pertaining to buckling of component supports. With respect to buckling, we find the applicant's criteria acceptable. As previously discussed, the allowable stress limits for support bolting of Class 1, 2, and 3 components should be provided.

The applicant states that "For the NSSS scope of supply, no valve operators which are mounted on Class 1 piping will be used as component supports.". Are any valve operators mounted on ASME Class 2 and 3 or ANSI B31.1 piping used as component supports? If so, provide a listing of these and an example of the analysis done. Similar information is also required for the BOP scope of responsibility.

Not enough detail is provided on the design and testing of snubbers. Do the design loads on the snubbers include those from SRV discharge and the LOCA? What are the criteria used for the snubber tests? A description of the actual tests are also required.

Into what category are the stresses due to differential anchor support movements placed for supports in the BOP scope of responsibility.

What elastic/plastic analysis has been done on supports? Provide an example of a typical analysis.

Based on our review of FSAR section 3.9.3.4 and contingent upon resolution of the open items, our findings will be as follows:

The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that, in the event of an earthquake 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 support components to withstand the most adverse combination of loading events without loss of structural integrity or supported component operability. The design and service load combinations and associated stress and deformation limits specified by ASME Code Class 1, 2, and 3 component supports comply with Standard Review Plan Section 3.9.3 and satisfy the applicable portions of General Design Criteria 1, 2, and 4.

3.9.4 Control Rod Drive Systems

Our review under Standard Review Plan Section 3.9.4 covers the design of the hydraulic control rod drive system up to its interface with the control rods. We reviewed the analyses and tests performed to assure the structural integrity and operability of this system during normal operation and under accident conditions. We also reviewed the life-cycle testing performed to demonstrate the reliability of the control rod drive system over its 40-year life.

The applicant has made reference to allowable deformations but they are not defined or listed. This must be included in the FSAR.

Based upon our review of FSAR Section 3.9.4 and contingent upon the satisfactory resolution of the open items, our findings are as follows:

The design criteria and the testing program conducted in verification of the mechanical operability and life cycle capabilities of the conrol rod drive system are in conformance with Standard Review Plan Section 3.9.4. The use of these criteria provide reasonable assurance that the system will function reliably when required and will form an acceptable basis for satisfying the mechanical reliability requirements of General Design Criterion 27.

3.9.5 Reactor Pressure Vessel Internals

Our review under Standard Review Plan Section 3.9.5 is concerned with the load combinations, allowable stress limits, and other criteria used in the design of the Perry reactor internals. The applicant has stated that the reactor internals have been designed in accordance with Subsection NG, "Core Support Structures", of the ASME Code, Section III. The description of the configuration and general arrangement of the reactor internal structures, components, assemblies and systems has been reviewed and found to be quite complete.

What feedwater sparger design is used at Perry? The applicant should provide a commitment to NUREG-0619.

Have the reactor internals placed in the "other internals" category been seismically analyzed to show that they will not compromise the integrity of seismically qualified reactor internals during the SSE.

Based upon our review of FSAR Section 3.9.5 and contingent upon the satisfactory resolution of the open items, our findings will be as follows:

The specified transients, design and service loadings, and combination of loadings as applied to the design of the Perry reactor internals provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these reactor internals would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these reactor internals to withstand the most adverse loading events which have been postulated to occur during the service lifetime without loss of structural integrity or impairment of function. The design procedures and criteria used by the applicant in the design of the Perry reactor internals comply with Standard Review Plan Section 3.9.5 and constitute an acceptable basis for satisfying the applicable requirements of General Design Criteria 1, 2, 4 and 10.

3.9.6 Inservice Testing of Pumps and Valves

In Sections 3.9.2 and 3.9.3 of this Safety Evaluation Report we discussed the design of safety-related pumps and valves in the Perry facility. The design of these pumps and valves is intended to demonstrate that they will be capable of performing their safety function (open, close, start, etc.) at any time during the plant life. However, to provide added assurance of the reliability of these components, the applicants will periodically test all its safety-related pumps and valves. These tests are performed in general accordance with the rules of Section XI of the ASME Code. These tests verify that these pumps and valves operate successfully when called upon. Additionally, periodic measurements are made of various parameters and compared to baseline measurements in order to detect long-term degradation of the pump or valve performance. Our review under Standard Review Plan Section 3.9.6 covers the applicant's program for preservice and inservice testing of pumps and valves. We give particular attention to those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code.

The applicant must provide a commitment that the inservice testing of ASME Class 1, 2, and 3 components will be in accordance with the revised rules of 10 CFR, Part 50, Section 50.55a, paragraph (g).

The applicant has not yet submitted its program for the preservice and inservice testing of pumps and valves; therefore, we have not yet completed our review.

Any requests for relief from ASME Section XI should be submitted as soon as possible.

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leak tight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems, thus causing the inner-system LOCA.

Pressure isolation valves are required to be category A or AC per IWV-2000 and to meet the appropriate requirements of IWV-3420 of Section XI of the ASME Code except as discussed below.

Limiting Conditions for Operation (LCC) are required to be added to the technical specifications which will require corrective action; i.e., shut-down or system siolation when the final approved leakage limits are not met. Also, surveillance requirements, which will state the acceptable leak rate testing frequency, shall be provided in the technical specifications.

Periodic leak testing of each pressure isolation valve is required to be performed at least once per each refueling outage, after valve maintenance prior to return to service, and for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position unless justification is given. The testing should also be performed after all disturbances to the valves are complete, prior to reaching power operation following a refueling outage, maintenance, etc.

The staff's present position on leak rate limiting conditions for operation must be equal to or less than 1 gallon per minute for each valve (GPM) to ensure the integrity of the valve, demonstrate the adequacy of the redundant pressure isolation function and give an indication of valve degradation over a finite period of time. Significant increases over this limiting value would be an indication of valve degradation from one test to another.

Leak rates higher than 1 GPM will be considered if the leak rate changes are below 1 GPM above the previous test leak rate or system design precludes measuring 1 GPM with sufficient accuracy. These items will be reviewed on a case by base basis.

The Class 1 to Class 2 boundary will be considered the isolation point which must be protected by redundant isolation valves.

In cases where pressure isolation is provided by two valves, both will be independently leak tested. When three or more valves provide isolation, only two of the valves need to be leak tested.

Provide a list of all pressure isolation valves included in your testing program along with four sets of Piping and Instrument Diagrams which describe your reactor coolant system pressure isolation valves. Also discuss in detail how your leak testing program will conform to the above staff position.

We will report the resolution of these issues in a supplement to the Safety Evaluation Report.

REACTOR

JAN 2 5 1982

4.3 Nuclear Design

The staff review of the nuclear design was based on information supplied by the applicant, in the FSAR and the referenced topical reports. The staff review was conducted within the guidelines provided by SRP Section 4.3.

4.3.1 Design Bases

Design bases are presented which comply with the applicable GDC. Acceptable fuel design limits are specified (GDC 10), a negative prompt feedback coefficient is specified (GDC 11), and tendency toward divergent operation (power oscillation) is not permitted (GDC 12). Design bases are presented which require a control and monitoring system (GDC 13) that automatically initiates a rapid reactivity insertion to prevent exceeding fuel design limits in normal operation or anticipated transients (GDC 20). The control system is required to be designed so that a single malfunction or single operator error will cause no violation of fuel design limits (GDC 25). A standby liquid control system is provided which is capable of bringing the reactor to cold shutdown conditions (GDC 26), and the control system is required to control reactivity changes during accident conditions when combined with the engineered safety features (GDC 27). Reactivity accident conditions are required to be limited so that no damage to the reactor coolant system boundary occurs (GDC 28).

The staff finds the design bases presented in the FSAR acceptable.

4.3.2 Design Description

Descriptions of the first cycle enrichment distribution, burnable poison loading, plutonium buildup, delayed neutron fraction, neutron lifetime, and core burnup have been provided. The values presented for these parameters are consistent with the design bases and are acceptable.

Power Distribution

The staff has reviewed the methods used by GE to predict power distributions during core lifetime (see Section 4.3.3 below). These methods have been compared to measured power distributions in operating BWRs to demonstrate their acceptability. Power distributions are controlled during reactor operation by adherence to predetermined control rod sequences so as to limit the maximum heat generation rate and minimum critical power ratio to values specified in the Technical Specifications. Power distributions will be monitored during reactor operation by the incore detector system. This system, described in the GE Topical Report APED-5076, "Incore Monitoring System for General Electric Water Reactors," consists of a source range monitoring subsystem (up to 10^{-5} full power), an intermediate range monitoring subsystem (10^{-6} to 0.2 full power), and an average power range monitoring subsystem (~0.05 - 1.5 full power). In addition a traversing incore probe (TIP) subsystem is used to calibrate the local power range monitors and to obtain detailed axial power distributions.

The intermediate range and average power range monitoring subsystems are each equipped with trips to scram the reactor if core flux levels reach undesirable values. These systems satisfy GDC 13 with respect to neutron flux monitoring. A comparison of calculated and measured power distributions is given in NEDO-20946, "BWR Simulator Verification Methods" (see Section 4.3.3 below). This comparison demonstrates that GE design methods are capable of adequately representing reactor operating states.

The staff concludes that discussions of power distributions in Section 4.3 and in the other documents referenced above are acceptable. The staff further concludes that the information presented concerning monitoring of power distributions presented in the FSAR and in Topical Report NEDO-20340, "Process Computer Performance Evaluation Accuracy," is acceptable.

Reactivity Coefficients

The most significant reactivity coefficients with respect to the stability and dynamic behavior of the reactor are the void coefficient and the Doppler coefficient. Of less significance is the moderator temperature coefficient. The fuel temperature, or Doppler, coefficient of reactivity will be negative at all operating conditions and times in life. The moderator void coefficient will also be always negative. The presence of negative Doppler and void coefficients during all operating conditions satisfies GDC 11 which requires a negative prompt reactivity coefficient in the power operating range. The moderator temperature coefficient may become slightly positive for certain operating conditions (very low power at end of life), but its effect is overshadowed by that of the other coefficients. GE has submitted a Topical Report, NEDO-20964, "Generation of Void and Doppler Reactivity Feedback for Application to BWR Design," which describes the methods used to obtain void and Doppler Reactivity coefficients. This report is currently under review (see Section 4.3.3 below), and the staff concludes, based on the review to date, that predictions of the various reactivity coefficients are suitably performed.

The calculated values of void and Doppler coefficients are multiplied by design conservatism factors and used in a point kinetics neutronics model to calculate the results of plant transients. Comparison of such calculations with experiments (for example, the Peach Bottom 2 turbine trip tests) have shown them to be nonconservative for certain core-wide transients involving overpressurization and rapid void collapse. Accordingly, the staff requires that such transients be analyzed by the recently approved ODYN code, which is described in NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volumes 1, 2, and 3. This code has been verified by comparison with the Peach Bottom 2 tests and others. The staff concludes that the ODYN code provides an acceptable calculation of pressurization transients when used in the manner prescribed in the letter dated February 4, 1981 from R. Tedesco (NRC) to G. G. Sherwood (GE). The point kinetics code is acceptable for other transients when the design conservatism factors are used.

Control Requirements

To allow for changes in reactivity due to reactor heatup (fuel and moderator temperature rise and void formation), load following (transient xenon), equilibrium xenon and samarium, and fuel burnup with consequent fission product buildup, a significant amount of excess reactivity is built into the core at beginning of life. In BWRs this excess reactivity is accounted for by the control rods except for a portion of that needed to account for fuel burnup. That portion is accounted for by burnable poison located in the fuel assemblies. The burnable poison also functions to shape the radial and axial power.

The applicant has presented data to show that sufficient control exists to satisfy the above requirements with enough additional control to provide a cold xenon-free effective multiplication factor ≤ 0.99 at the most reactive point in the core lifetime with the rod having the highest worth stuck out of the core. NEDO-29046 provides comparisons of calculated and measured cold critical states and provides a demonstration that calculation of shutdown margins is adequate. The staff concludes that suitably conservative assessments of reactivity control requirements have been made and that adequate reactivity control has been provided to ensure shutdown capability, even with one rod stuck out of the core.

A standby liquid control system is provided which is completly independent of the control rod system and is capable of shutting down the reactor and maintaining it in the cold shutdown state at any time in core life. This satisfies the shutdown reactivity requirement of GDC 26.

Control Rod Patterns and Reactivity Worth

Startup and operation of the reactor will be performed by manipulation of control rods and control of recirculation flow. The control rods will be withdrawn in sequence according to predetermined patterns. These patterns are established in such a way that the following design criteria are met:

(1) Control rod worths shall be limited so as to have acceptable consequences if a rod is dropped from the fully inserted to a withdrawn position (rod drop accident).

(2) Control rod withdrawal increments of rod worth shall be limited so that withdrawal of a control rod by one notch does not produce a period that cannot be handled by the operator.

(3) In BWR-6 reactors, certain of the rods may be withdrawn in gangs of three or four rods at a time. Analyses have been done to show that the above design criteria are met for such withdrawals.

During operation, the rod patterns are monitored by the rod pattern control system function of the rod control and information system up to the preset power level (~20 percent power). Above this power level, rod worths are not sufficient to violate the above criteria. The rods are withdrawn in the banked position withdrawal sequence which is described in Topical Report NEDO-21231, "Banked Position Withdrawal Sequence." This report has been reviewed and accepted by the staff, and the staff concludes that this operating mode is acceptable for use in the Perry reactor.

Stability

The stability of large boiling water reactors to xenon oscillations has been discussed in the GE Topical Report APED-5640 "Xenon Considerations in Design of Large Boiling Waters Reactors." These studies show that a BWR will be stable to any xenon-induced power oscillation because of the damping effect of the large, negative, spatially varying void coefficient. In addition, attempts to induce undamped xenon oscillations in operating BWRs confirm the presence of a large damping effect.

The staff concurs with the conclusion that large BWRs will be stable to xenon-induced power oscillations.

This satisfies part of GDC 12, which requires that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. Thermal-hydraulic stability is discussed in Section 4.4 below.

Criticality of Fuel Assemblies

The staff has reviewed the information presented regarding criticality of fuel assemblies in storage and handling operations. Calculations are performed for higher-than-normal enrichment fuel assemblies, and no credit is taken for burnable poison in the fuel. The K_{eff} value for two bundles side by side in a pool is ~0.74. Four bundles in a square arrangement have a K_{eff} value of ~0.90. An array of 12-14 fuel bundles is required for criticality if no gadolinia is assumed to be present. An array of 16 to 20 fresh fuel bundles are required for criticality if gadolinia is assumed to be present.

Vessel Irradiation

Neutron fluences at the inside surface of the pressure vessel have been calculated using a one-dimensional discrete ordinates code in an infinite cylinder geometry. Design radial and axial power distributions are used and the calculation is performed with the axially peaked value of the flux. An azimuthal peaking factor is applied to the calculated flux to obtain the peak in the azimuthal direction. Operation at full power for 40 years with an 80 percent utilization factor was assumed. Anisotropic scattering effects were included outside the core and the resultant fluence was 4.3×10^{18} neutrons per cm^2 for neutrons having energies greater than one million electron-volts.

The staff concludes that the calculation method used to obtain the vessel fluence is state of the art and is acceptable. The staff further concludes that the core and externals have been properly modelled and that suitable values of vessel fluence have been obtained.

4.3.3 Analytical Methods

The staff reviewed and evaluated the information presented on the analytical methods. The basic calculational procedures which are used by GE for generating neutron cross sections are part of its so-called Lattice Physics Model. In this model, the neutron spectrum is divided into energy regions and a different technique is used in each region. The fast groups (fission source energy range) are treated by multigroup integral collision probabilities to account for geometrical effects in fast fission. Epithermal cross sections are calculated by the B1 method of the widely used GAM code. Resonance energy cross sections for this energy range are calculated by using the intermediate resonance approximation with energy- and position-dependent Dancoff factors included. The thermal cross sections are computed by a THERMOS-type program.

This program accounts for the spatially varying thermal spectrum throughout a fuel bundle. These calculations are performed for an extensive combination of parameters including fuel enrichment and distribution, fuel and moderator temperatures, burnup, voids, void history, the presence or absence of adjacent control rods, and gadolinia concentration and distribution in the fuel rods. As part of the Lattice Physics Model, three-group two-dimensional XY diffusion calculations for one or four fuel bundles are performed. In this way, local fuel rod powers can be calculated as well as single bundle or four bundle (with or without a control rod present) average cross sections.

GE has submitted a licensing topical report NEDE-20913-P, "Lattice Physics Methods," which describes in detail the procedures outlined above. The staff has reviewed this report and has concluded that the discussion therein permits a knowledgeable person to conclude the methods employed are state of the art. The staff further concluded that the methods satisfy the provisions of the SRP for core physics methods and are acceptable.

The single- or four-bundle average neutron cross sections which are obtained from the Lattice Physics Model are used in either two- or three-dimensional diffusion calculations. Two-dimensional XY calculations are usually performed in three groups at a given axial location to obtain gross power distribution, and reactivities. The three-dimensional diffusion calculations use one energy group and can couple neutronic and thermal-hydraulic phenomena. These three-dimensional calculations are performed using 24 axial nodes and one radial node per fuel bundle. This three-dimensional calculation provides power distributions, void distributions, control rod positions, reactivities, eigenvalues, and average cross sections for use in the one-dimensional axial calculations. The three-dimensional calculations have been described in a topical report, NEDO-20953, "Three Dimensional BWR Core Simulator". The staff has reviewed this report and has reached similar conclusions to those reached for the Lattice Physics Methods report.

The one-dimensional calculation referred to above is a space-time diffusion calculation which is coupled to a single channel thermal-hydraulic model. This axial calculation is used to generate the scram reactivity function for various core operating states. This one-dimensional space-time code has been compared by GE with results obtained using the industry standard code, WIGLE, and shown to be conservative. The staff consultant, Brookhaven National Laboratories, has performed an extensive study of BWR scram reactivity behavior (BNL-NUREG-50584, "A Dynamic Analysis of BWR Scram Reactivity Characteristics") and has concluded that the end of cycle all rods out configuration represents the limiting condition for BWR scram system effectiveness. Thus, the staff concludes that the method and assumptions used by GE to obtain the scram reactivity curve are acceptable. (See discussion under Reactivity Coefficients above for application of the scram curve.)

The Doppler, moderator void, and moderator temperature reactivity coefficients are generated in a straightforward manner from data obtained from the Lattice Physics Model. The effective delayed neutron fraction and the prompt mode neutron lifetime are computed using the one-dimensional space-time code. The power coefficient is obtained by appropriately combining the moderator void, Doppler, and moderator temperature reactivity coefficients.

GE has submitted a licensing topical report, NEDO-20964, "Generation of Void and Doppler Reactivity Feedback for Application to BWR Design." The staff is currently reviewing this report. Based on the review to date, the staff does not anticipate any major changes to the methods. (See comment in Section 4.3.2 above.)

The effect of spatially varying xenon concentrations on the stability of a boiling water reactor is specifically discussed in GE Topical Report APED-5640. These studies show that a BWR will be stable to any xenon-induced power oscillations because of the damping effect of the large, negative, spatially varying void coefficient.

Comparisions between calculated and measured local and gross power distributions have been presented by the GE in two topical reports, NEDO-20939, "Lattice Physics Methods Verification," and NEDO-20946, "BWR Simulator Methods Verification." Local (intra-bundle) power distribution comparisons were made to data obtained from critical experiments and from gamma scans performed on operating plants. Gross radial and axial power distributions obtained from operating plants have been compared with values predicted by the BWR simulator code. These comparisons have yielded values for calculational uncertainties to be applied to power distributions. Comparisons have also been made of calculated values of cold, xenon-free reactivity and hot operating reactivity of a number of operating reactors as a function of cycle exposure. These comparisons have been used to establish shutdown reactivity requirements.

The staff has reviewed the two topical reports, NEDO-20939 and NEDO-20946, and found them acceptable for reference in licensing actions.

4.3.4 Evaluation Finding

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. The staff concludes that the information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the Perry plant.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a signficant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity requirements for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to shut down the reactor with at least a 1.0 percent $\Delta k/k$ subcritical margin in the cold condition at any time during the cycle with the highest worth control rod stuck in the fully withdrawn position.

On the basis of its review, the staff concludes that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to ensure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available.

The staff concludes that the nuclear design is acceptable and meets the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28. This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 11 with respect to prompt inherent nuclear feedback characteristics in the power operating range by

- calculating a negative Doppler coefficient of reactivity

- using calculational methods that have been found acceptable

The staff has reviewed the Doppler reactivity coefficients in this case and found them to be suitably conservative.

(2) The applicant has met the requirements of GDC 12 with respect to power oscillations that could result in conditions exceeding specified acceptable fuel design limits by:

- showing that such power oscillations are not possible and/or can be easily detected and thereby remedied

- using calculational methods that have been found acceptable

The staff has reviewed the analysis of these power oscillations in this case and found them to be suitably conservative.

(3) The applicant has met the requirements of GDC 13 with respect to provision of instrumentation and controls to monitor variables and systems that can affect the fission process by

- providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure

- providing suitable alarms and/or control room indications for these monitored variables

2. The applicant has met the requirements of GDC 12 with respect to power oscillations which could result in conditions exceeding specified acceptable fuel design limits by:

- showing that such power oscillations are not possible and/or can be easily detected and thereby remedied, and

- using calculational methods that have been found acceptable

The staff has reviewed the analysis of these power oscillations in this case and found them to be suitably conservative.

(3) The applicant has met the requirements of GDC 13 with respect to provision of instrumentation and controls to monitor variables and systems that can affect the fission process by

- providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure

- providing suitable alarms and/or control room indications for these monitored variables

(4) The applicant has met the requirements of GDC 26 with respect to provision of two independent reactivity control systems of different designs by

- having a system than can reliably control anticipated operational occurrences

- having a system that can hold the core subcritical under cold conditions

- having a system that can control planned, normal power changes

(5) The applicant has met the requirements of GDC 27 with respect to reactivity control systems that have a combined capability in conjunction with action by the emergency core cooling system reliably controlling reactivity changes under postulated accident conditions by

- providing a movable control rod system and a liquid poison system
- performing calculations to demonstrate that the core has sufficient shutdown margin with the highest worth stuck rod

(6) The applicant has met the requirements of GDC 28 with respect to postulated reactivity accidents by

- meeting the fuel enthalpy limit of 280 cal/gm
- meeting the criteria on the capability to cool the core
- using calculational methods that have been found acceptable for reactivity insertion accidents

(7) The applicant has met the requirements of GDC 10, 20, and 25 with respect to specified acceptable fuel design limits by providing analyses demonstrating

- that normal operation, including the effects of anticipated operational occurrences, have met fuel design criteria
- that the automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and ensures the automatic operation of systems and components important to safety under accident conditions
- that no single malfunction of the reactivity control system causes violation of the fuel design limits

4.4 Thermal Hydraulic Design

4.4.1 Thermal Hydraulic Design Bases

The thermal-hydraulic safety design bases for Perry Units 1 and 2 can be summarized as follow:

(1) No fuel damage occurs as a result of moderate frequency transient events. Specifically, the minimum critical power ratio (MCPR) operating limit is specified such that at least 99.9 percent of the fuel rods in the core are not expected to experience boiling transition during the most severe moderate frequency transient events.

(2) The core and fuel design basis for steady-state operation, i.e., MCPR and LHGR limits, have been defined to provide margin between steady-state operating conditions and any fuel damage condition to accommodate uncertainties and to assure that no fuel damage results even during the worst moderate frequency transient condition at anytime in life.

(3) No undamped oscillations or other hydraulic instabilities should occur for normal operation nor for the most severe moderate frequency transient event.

4.4.2 Thermal-Hydraulic Design Methodology

4.4.2.1 Critical Power Ratio (CPR) Correlation

The occurrence of boiling transition is a function of the local steam quality, boiling length, mass flow rate, pressure, flow geometry, and local peaking pattern. The (critical) quality at which boiling transition occurs as a function of the distance from the equilibrium boiling boundary is predicted by the GEXL (General Electric Critical Quality, X_c - Boiling Length) correlation. The GEXL correlation is described in the General Electric Company Topical Reports, NEDO-10958A, "General Electric Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application," January 1977 and NEDE-24196, "Basis for BWR 6 8 x 8 Fuel Thermal Analysis Application," December 1979. The figure of merit used for reactor design and operation is the critical power ratio, the ratio of the critical bundle power to the operating bundle power. This correlation and supporting topical reports have been reviewed and approved by the staff.

Critical power tests have been run on prototypical 8 x 8 fuel bundles with two water rods. Test data for cosine axial heat flux shapes indicate that the water rods do not affect the GEXL capability of predicting the bundle critical power performance for bundle radial peaking patterns typical of 8 x 8 retrofit fuel. We have previously found that the GEXL data base, which includes top and bottom peaked axial heat flux distributions, combined with the two water rod data demonstrate the adequacy of the GEXL correlation to predict critical power in both 8 x 8 and 8 x 8 retrofit bundles. We have previously concluded that the GEXL correlation is acceptable for both 8 x 8 and 8 x 8 retrofit fuel application.

4.4.2.2 Thermal Hydraulic Stability

Recent BWR fuel design changes, decreasing the fuel rod size and increasing the gap conductance (due to prepressurization of the fuel), are reducing BWR stability margins. The maximum decay ratio for most BWRs has increased and now exceeds 0.5, which has been the General Electric design criteria for BWR stability. GE now proposes a decay ratio of 1.0 for their thermal hydraulic stability criteria.

The Perry stability analysis resulted in a maximum decay ratio of 0.97 for end-of-life cycle. The staff has not approved the proposed stability criterion of a 1.0 decay ratio or the FABLE Code used to calculate the decay ratio. This code is described in General Electric Company Topical Report,

NEDO-21506, "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," January 1977.

To further evaluate this criterion and other stability criteria, we are performing a generic study of the hydrodynamic stability characteristics of light water reactors under normal operation, anticipated transients, and accident conditions. The results of this study will be applied to our review and acceptance of stability analyses and analytical methods now in use by the reactor vendors. The staff has previously approved for operation cycle 1 core designs having calculated maximum decay ratio values as high as 0.7 during initial cycle. We have not accepted the analyses for later cycles which have higher calculated decay ratios, comparable to the 0.97 value calculated for Perry.

While we continue our generic evaluation, we have concluded that the Perry core design stability will be acceptable for cycle 1 only.

However, in order to provide additional margin to stability limits, natural circulation operation of Perry Units 1 and 2 will be prohibited until our review of these conditions is completed. Any action resulting from our study will be applied to Perry Units 1 and 2. The applicant has indicated in response to our question that they will accept the resolution of this issue done by Licensing Review Group II (LRG-II) and the staff. This is acceptable to the staff. In addition, we will condition the operating license to require that a new stability analysis be submitted and approved prior to second cycle operation.

4.4.3 Design Abnormalities

4.4.3.1 Crud Deposition

Crud deposition causes flow reduction in some light water reactor cores. However, measurement of core flow by jet pump pressure drop and core plate pressure drop will provide adequate indication of flow reduction, if it should occur. Technical Specifications will be modified to require that the core flow be checked at least once every 24 hours to verify that the core flow is consistent with the rated flow along the power/flow control line. This frequency is sufficient to detect crud deposition effects. For pressure drop considerations in design analysis, it is assumed that a conservative amount of crud is deposited on the fuel rods and the fuel rod spacers. This is reflected in a decreased flow area, increased friction factors, and increased spacer loss coefficients. The effect of this crud deposition is to increase the core pressure drop by approximately 1.7 psi. We conclude that the assumptions regarding crud deposition used in design analyses in conjunction with the required flow monitoring are acceptable.

4.4.4 Loose Parts Monitoring

The applicant has provided a description of the Loose Parts Monitoring System (LPMS) which will be used by Perry Units 1 and 2. The design will include two sensors at each selected natural collection region. Twelve sensing channels (6 channel pairs) will be provided to detect a loose part that will weigh 0.25-30 lbs and will impact with kinetic energy of 0.5 ft-lb within 3 ft of each sensor. In response to a question, the applicant has committed to follow the recommendations of Regulatory Guide 1.133, Revision 1 (May 1981). Alarm settings will be established based on the baseline data taken during startup testing at selected nominal power levels. The staff has evaluated the Perry Units 1 and 2 LPMS by comparing it with the equipment and procedures used on other comparable plants, taking into account pertinent differences. In addition the applicant has provided a detailed discussion of the operator training program for operation of the LPMS, planned operating procedures and record keeping procedures. We will require that location of the required sensors, limiting conditions for operation, and surveillance requirements be included in the Technical Specifications in accordance with Regulatory Guide 1.133, Revision 1 (May 1981). We will require that the notification of loose parts be done as summarized in Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A Technical Specifications. Based on our review and the design and submittal commitments made by the applicant, the staff concludes that an acceptable LPMS and program will be implemented for Perry Units 1 and 2.

4.4.5 Thermal-Hydraulic Models

Steady-state thermal hydraulic analyses was performed using the multichannel thermal-hydraulic model ISCOR computer code. The ISCOR code and another GE Code PANACEA (3 dimensional BWR core simulator) use the same steady state thermal hydraulic model as described in NEDO-20953A, dated January 1977. The staff finds this description is inadequate. In response to our question, the applicant has committed to supply proper documentation of the ISCOR code. The steady-state operation MCPR limit originally proposed was based upon calculations using the REDY model described in General Electric Topical Report, NEDO-10802. The results from tests performed at Peach Bottom-2 revealed that in certain cases the results predicted by REDY are nonconservative. Therefore, we are requiring the applicant to use the ODYN code to analyze the following transients: (1) feedwater controller failure - maximum demand, (2) generator load rejection, and (3) turbine trip with and without bypass. The ODYN code is described in General Electric Topical Reports NEDO-24154 and NEDE-24154-P and has been previously accepted by the staff.

The MCPR during significant abnormal events is calculated using SCAT code, a transient core heat transfer analysis code. This code is based on a multinode, single channel thermal-hydraulic model which requires simultaneous solution of partial differential equations for the conservation of mass, energy, and momentum in the bundle, and which accounts for axial variation in power generation. The primary inputs to the model include a physical description of the bundle, and channel inlet flow and enthalpy, pressure and power generation as functions of time. This code is described in detail in General Electric Topical Report, NEDO-20566. In response to our question, the applicant has agreed that they have used modified SCAT code in conjunction with ODYN code for GETAB MCPR evaluation of all the transients. The applicant has committed to supply the proper documentation of this code. Upon receipt of proper documentation from the applicant, we will verify that these codes are acceptable and that the applicant has calculated correctly the steady-state operating limit for the minimum critical power ratio.

4.4.6 Thermal-Hydraulic Comparison

A summary of the thermal-hydraulic parameters for Perry Units 1 and 2 is given in Table 4.4-1. A comparison with the parameters for Hatch-2 and Grand Gulf core designs are given for reference. The Hatch-2 core design has been previously approved in the Safety Evaluation Report issued in June 1978 and Hatch-2 is an operating reactor. The Grand Gulf Core design has been previously approved in the Safety Evaluation Report issued in August 1981.

The primary difference in core design between Perry Units 1 and 2 and Grand Gulf Units 1 and 2 and Hatch-2 is size. Perry Units 1 and 2 have 748 bundles compared to 800 bundles for Grand Gulf and 560 bundles for Hatch-2. Perry and Grand Gulf have higher power density compared to Hatch-2. The design thermal power for Perry is 3579 MWt compared to 3833 MWt for Grand Gulf and 2436 MWt for Hatch-2. The average orifice pressure drop for Perry as compared to Grand Gulf and Hatch-2 is lowest in the central region of the core (5.71 psia vs. 5.78 psia vs. 8.0 psia) and is higher than Hatch-2 and lower than Grand Gulf in the peripheral region of the core (18.68 psia vs. 19.16 psia vs. 16.52 psia). Hatch-2 is a BWR/4 core and Perry and Grand Gulf are BWR/6 cores. All use the prepressurized 8 x 8 retrofit (P8 x 8R) fuel assemblies. The Perry thermal hydraulic design is comparable to that of Grand Gulf and Hatch-2.

4.4.7 Single Loop Operation

Since no analysis has been presented for minimum critical power ratio limits or stability characteristics for single loop operation, we will require by Technical Specifications that single loop operation will not be permitted until supporting analyses are provided and approved.

4.4.8 Conclusions and Summary

The thermal-hydraulic design of the core for Perry has been reviewed. The acceptance criteria used as the basis for our evaluation are set forth in the Standard Review Plan (SRP), NUREG-0800 in 4.4, Section II, "Thermal and Hydraulic Design Acceptance Criteria." The scope of our review included the design criteria, core design, and the steady-state analysis of the core thermal-hydraulic performance. The review concentrated on the differences between the proposed core design and those designs that have been previously reviewed and found acceptable by the staff. It was found that all such differences were acceptable. Perry's thermal-hydraulic analyses were performed using appropriate methods and correlations and this is acceptable to the staff.

The staff concludes that the initial core has been designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded during

Table 4.4-1 Thermal-hydraulic design parameters

	Hatch-2 (218-560)	Grand Gulf 1 & 2 (251-800)	Perry 1 & 2 (238-748)
Design thermal output - MWt	2,436	3,833	3,579
Final feedwater temperature (FFWT) - °F	420	420	420
Steam flow rate at FFWT - 106 lbs/hr	10.47	16.49	15.4
Core coolant flow rate - 106 lb/hr	77.0	112.5	104.0
Feedwater flow rate - 106 lb/hr	10.44	16.46	15.367
Steam pressure, nominal in steam down - psia	1,020	1,040	1,040
Steam pressure, nominal core design - psia	1,035	1,055	1,055
Average power density - kw/liter	49.15	54.1	54.1
Maximum linear thermal output - kw/ft	13.4	13.4	13.4
Average linear thermal output - kw/ft	5.38	5.9	5.9
Core total heat transfer area - ft2	54,879	78,398	73,303
Fuel type	P8 x 8R	P8 x 8R	P8 x 8R
Water rods per bundle	2	2	2
Core inlet enthalpy at FFWT - Btu/lb	526.9	527.9	527.7
Core maximum exit void within assemblies - %	76.3	76.0	76.0
Core average void, active coolant - %	42.2	41.0	41.4
Active coolant flow area per assembly, in2	15.824	15.164	15.164
Core average inlet velocity, ft/sec	6.6	7.07	6.98
Total core pressure drop - psia	23.9	26.74	26.40
Core support plate pressure drop - psia	19.46	22.32	22.0
Average orifice pressure drop psia			
Central region	8.0	5.78	5.71
Peripheral region	16.52	19.16	18.68
Number of fuel rods per bundle	62	62	62
Rod outside diameter - in			
Fuel rod	0.483	0.483	0.483
Water rod	0.591	0.591	0.591
Active fuel length - in	150	150	150
Rod pitch - in	0.640	0.640	0.640

steady-state operation and anticipated operational occurrences. The thermal-hydraulic design of the initial core, therefore, meets the requirements of General Design Criterion 10, 10 CFR Part 50, and is acceptable for preliminary design approval. This conclusion is based on the applicant's analyses of the core thermal-hydraulic performance which were reviewed by the staff and found to be acceptable.

The applicant has committed to a preoperational and initial start-up test program in accordance with Regulatory Guide 1.68 to measure and confirm thermal-hydraulic design aspects. The staff has reviewed the applicants preoperational and initial start-up test program and has concluded that it is acceptable. However, prior to final design approval and issuance of an operating license, the staff will require the applicant to resolve the following:

- Provide acceptable documentation of the ISCOR code used in the thermal-hydraulic analyses and modified SCAT Code (for use with ODYN Code) used in all the GETAB-MCPR evalution of the transients for Perry.

- Provide by separate amendment, the operating limit MCPR as calculated by including the ODYN methods.

The following items need to be addressed in the Technical Specifications:

- Single loop operation is not permitted unless supporting analyses are provided and approved;

- Operation in a natural circulation mode is not permitted while we continue our generic evaluation of thermal-hydraulic stability for BWRs; and

- The core flow should be checked at least once every 24 hours to account for possible effects of crud deposition.

In addition, a license condition will be imposed as follows:

- Operation beyond Cycle 1 is not permitted until stability analysis is provided and approved.

The applicant's response to Item II.F.2, "The Need for Additional Instrumentation for Detection of Inadequate Core Cooling (ICC)," of NUREG-0737, "Clarification of TMI Action Plan Requirements," has recently been received. The applicant is a member of the BWR owner's Licensing Review Group II (LRG-II). The staff is pursuing with LRG-II group the need for additional instrumentation needed for detection of ICC. The staff's conclusions concerning this issue will be applied to Perry Nuclear Power Plant.

These issues will be addressed in the input to the final SER for Perry FSAR.

4.6 Functional Design of Reactivity Control Systems

The control rod drive system (CRDS) and recirculation flow control system (RFCS) are designed to control reactivity during power operation. Reactivity is controlled in the event of fast transients by automatic rod insertion. In the event the reactor cannot be shut down with the control rods, the operator can actuate the standby liquid control system which pumps a solution of sodium pentaborate into the primary system. The evaluation of the functional design of the standby liquid control system can be found in Section 9.3.5 of this report.

Reactivity in the core is controlled by the CRDS via movable control rods interspersed throughout the core. These rods control the overall reactor power level and provide the principal means of quickly and safely shutting down the reactor. This is the normal method of making large changes in reactor power, such as daily or weekly load shifts requiring reductions and increases of power.

Each control rod is moved by a separate hydraulic control unit. A supply pump provides the hydraulic control units with water from the condensate storage tank for cooling the rods and for moving them into and out of the core, with a spare pump on standby. The pump also provides water to a scram accumulator in each hydraulic control unit to maintain the desired water inventory. When necessary, the accumulator forces water into the drive system to scram the control rod connected to that hydraulic control unit; at lower pressures the volume of water in the scram accumulator is sufficient to scram the rod. At higher pressures, most of the water to scram is provided from the reactor vessel. A single failure in a hydraulic control unit would result in the failure of only one rod.

The CRDS has been designed to permit periodic functional testing during power operation with the capability to test individual scram channels and motion of individual control rods independently. The CRDS is designed so that failure of all electrical power will cause the control rods to scram, thereby protecting the reactor. Based on the above, we conclude that the requirements of General Design Criterion 23, "Protection System Failure Mode" are satisfied.

Preoperational tests of the control rod drive hydraulic system will be conducted to determine capability of the system. Start-up tests will be conducted over the range of temperatures and pressures from shutdown to operating conditions in order to determine compliance with applicable technical specifications. Each rod that is partially or fully withdrawn during operation will be exercised one notch at least once each week. Operable control rods will be tested for compliance with scram time criteria, from the fully withdrawn position, after each refueling shutdown.

A malfunction in the CRDS could result in a reactivity change. The applicant demonstrated in his safety analyses (Section 15 of the Final Safety Analysis Report) that the CRDS limits these postulated transients within acceptable fuel response, as required by General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunction."

The CRDS is designed to provide reactivity control under normal operation and anticipated operational occurrences with an appropriate allowance for a stuck rod. This capability is demonstrated by the safety analyses discussed in Section 15 of the Final Safety Analysis Report. This system is also capable of holding the core subcritical under cold shutdown conditions. The recirculation flow control system is capable of accommodating reactivity changes during normal operation conditions (i.e., power changes and xenon burnout). The standby liquid control system is capable of bringing the reactor subcritical under cold shutdown conditions in the event the control rods cannot be inserted. These systems, taken together, satisfy the requirements of General Design Criteria 26, "Reactivity Control System Redundancy and Capability," 27, "Combined Reactivity Control System Capability," and 29, "Protection Against Anticipated Operational Occurrences."

The CRDS is capable of providing reactivity control following postulated accidents with an appropriate margin for a stuck rod. This capability is demonstrated by the loss-of-coolant accident and rod dropout analyses presented by the applicant which, in turn, show that the consequences are acceptable and core cooling is maintained, as required by General Design Criterion 28, "Rectivity Limits."

The design does not utilize a CRDS return line to the reactor pressure vessel. In accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drives Return Line Nozzle Cracking," November 1980 equalizing valves are installed between the cooling water header and exhaust water header, the flow stabilizer loop is routed to the cooling water header, and both the exhaust header and flow stabilizer loop are stainless steel piping. The applicant has committed to conduct preoperational tests to verify the flow rate of the CRD hydraulic system to determine leakage (return) flow to the reactor vessel and to verify the proper operation of the CRDS. We find the CRD hydraulic system modification acceptable, pending satisfactory conclusion of the tests demonstrating adequate return flow to the reactor vessel (equal or greater than boiloff of the base case as noted in Section 8 of NUREG-0619 and thus satisfactory CRD operation.

We have reviewed the extent of conformance of the Scram Discharge Volume (SDV) design with the NRC generic study, "BWR Scram Discharge System Safety Evaluation," of December 1, 1980. The design provides two separate SDV headers, with an integral instrumented volume (IV) at the end of each header, thus providing close hydraulic coupling. Each IV has redundant and diverse level instrumentation for the scram function attached directly to the IV. Vent and drain lines are completely separated and contain redundant vent and drain valves equipped with redundant solenoid pilot valves. High point venting is provided. We conclude that the design of the SDV fully meets the requirements of the above referenced NRC generic SER and is therefore acceptable.

Based on our review, we conclude that the functional design of the reactivity control system meets the requirements of General Design Criteria 23, 25, 26, 27, 28 and 29 and the criteria of NUREG-0619 and the "BWR Scram Discharge System Safety Evaluation," dated December 1, 1980 with respect to demonstrating the ability to reliably control reactivity changes under normal operation, anticipated operational occurrences and accident conditions including single failures, and is therefore acceptable.

REACTOR COOLANT SYSTEM
AND CONNECTED SYSTEMS

SECTION 5

REACTOR COOLANT SYSTEM AND CONNECTING SYSTEMS
OPEN AREAS

Section 5.2.2-Overpressurization Protection --

1. The applicant must submit for our review and approval a plant overpressurization analysis using the ODYN code and including the effect of recirculating pump trip.

2. the applicant's technical specifications should include an initial operating pressure limit of 1045 psig for power and startup modes.

Section 5.2.5-Reactor coolant Pressure Boundary Leakage Detection --

1. The applicant is required to verify that the particulate channel of the fission products monitoring subsystem receives its power from a Class 1E source.

2. The applicant is required to provide assurance that steam from reactor-coolant leakage from sources such as the reactor vessel head flange vent drain and valve packings will be condensed for leak detection monitoring purposes. Additionally, the applicant is required to describe the surveillance program to minimize the potential for system blockage.

3. The applicant is required to describe how the operator will determine the amount of leakage by observing the indication available to him, including the need for unit conversion and how a record of background leakage is maintained. Additionally, if the monitoring is computerized, discuss the backup procedures that are to be provided assuming failure of the ~~xxx~~ computer.

4. The applicant is required to describe the proposed technical specifications that limit the conditions for identified and unidentified leakage and that address the availability of the various instrument types to assure adequate coverage all times.

Section 5.3.1 - Reactor Vessel Materials - Questions/open areas relative to this section are presented on the following page:

~~ENCLOSURE 2~~

~~REQUEST FOR ADDITIONAL INFORMATION~~

~~[illegible]~~

SECTION 5.3.1

123.0 MATERIALS ENGINEERING BRANCH--COMPONENT INTEGRITY SECTION

123.1 To demonstrate compliance with Paragraph III.B.3, certify that the calibration schedule for temperature instruments, drop weight, and Charpy V-notch machines comply with the requirements of Paragraph NB-2360 of the ASME Code. If they are not in compliance, indicate the schedule used and provide a basis for granting an exemption to the exact requirements of NB-2360 of the ASME Code.

123.2 To demonstrate compliance with the requirements of Paragraph IV.A.3 of Appendix G, 10 CFR Part 50, certify that all ferritic materials used for reactor coolant pressure boundary piping and valves which are in balance of plant and the nuclear steam supply system meet the requirements of NB-2300 of the ASME Code.

123.3 To demonstrate the surveillance capsule program complies with Paragraphs II.B and II.C of Appendix H.

(a) Provide a sketch showing the azimuthal location of each material surveillance capsule.

(b) Identify each plate specimen in each capsule by heat number and chemical composition, especially copper and phosphorus.

(c) Identify each weld specimen in each capsule by weld wire type and heat, flux type, lot identification, and chemical composition, especially copper and phosphorus.

(d) Identify the lead factor for each surveillance capsule.

ENCLOSURE 2
PERRY SAFETY EVALUATION

5.2.2 Overpressurization Protection

The reactor coolant pressure boundary (RCPB) is provided with a pressure relief system to:

(1) Prevent the pressure within the RCPB from rising beyond 110 percent of the design value, and

(2) Provide automatic depressurization for small breaks in the nuclear system occurring together with failure of the high pressure core spray system so that the low pressure coolant injection and the low pressure core spray systems can operate to protect the fuel barrier.

The relief system must permit verification of its operability, and withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, and faulted conditions.

Overpressurization protection at Perry is accomplished through the use of nineteen combination safety/relief valves of the Dikkers type mounted on the four main steam lines. The following table indicates that the Perry Pressure relief system design is similar to other BWR Class 4, 5, and 6 plants.

Plant	Class	Number of Pressure Relief Valves	Plant Rated Steam Flow (lb/hr)	Plant Rated Power (MWT)
Perry 1/2	6	19	1.54×10^7	3579
Grand Gulf 1/2	6	20	1.65×10^7	3833
Clinton 1/2	6	16	1.25	2894
LaSalle 1/2	5	18	1.42	3293
Susquehanna 1/2	4	16	1.35	3293
Fermi 2	4	15	1.42	3293
Shoreham	4	11	1.05	2436

All of the combination safety/relief valves discharge directly to the suppression pool. The valves are designed to meet seismic and quality standards consistent with the requirements of Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants," and RG 1.29, "Seismic Design Classification," as discussed in Section 3.2* of this report.

The basis for overpressure protection in a nuclear reactor is Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), General Design Criterion (GDC) 15, "Reactor Coolant System Design." This criterion requires that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. To satisfy this criterion, the overpressurization protection system for Perry was designed in compliance with the ASME Pressure Vessel Code - Section III which requires that the maximum pressure reached during the most severe pressure transient be less than 110 percent of the design pressure. For Perry this pressure limit is 1375 pounds per square inch gauge.

---

*As an aid to DL in the preparation of the SER, references have been made in this draft of Sections 5.2.2, 5.4.6, 5.4.7, 6.3, 15 and 22 to other sections that are written by branches other than RSB. The LPM should verify that the listed references are consistent with his SER section numbering system.

The nominal setpoints of the combination safety/relief valves are as follows:

Mode of Operation	Setpoint, psig Minimum	Maximum
Relief (power actuated)	1103	1123
Safety (spring setpoint)	1165	1190

Their total capacity at their set pressure is approximately 103% of rated steam flow. Prior to installation the safety relief valve manufacturer tests the valves hydrostatically for valve response, set pressure, and seat leakage to certify that design and performance requirements have been met. During the preoperational test program, specified manual and automatic actuation is verified in compliance with Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants." In addition, the applicant has stated that 50% of the valves will be removed for maintenance and inspection, and tested each refueling outage in accordance with Section XI of the ASME Boiler and Pressure Vessel Code per plant Technical Specification requirements.

It is 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 on any boiling water reactor (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. By letter dated November 5, 1981 from Dalwyn R. Davidson (The Cleveland Electric and Illuminating Company) to R. L. Tedesco (NRC) the applicant has indicated that a safety/relief valve surveillance program is being developed commensurate to that of the BWR Owners Group (LRG II), the primary objective of which is to gather data to identify generic safety/relief valve problems. It is further noted that the applicant is a participant in the BWR Owners Group program to test safety/relief valves in compliance with requirements of Item II.D.1 of NUREG-0737.

The applicant has analyzed a series of transients that would be expected to require pressure relief actuation to prevent overpressurization. These are tabulated below.

Pressurization Events Resulting In Pressure Relief Actuation

FSAR Subsection	Event
15.1.2	Feedwater Control Failure, Maximum Demand
15.1.3	Pressure Controller, Fail Open
15.2.2	Generator Load Rejection, Bypass-On
15.2.2	Generator Load Rejection, Bypass-Off
15.2.3	Turbine Trip, Bypass-On
15.2.3	Turbine Trip, Bypass-Off
15.2.4	Inadvertent MSIV Closure
15.2.5	Loss of Condenser Vacuum
15.2.6	Loss of Auxiliary Power Transformer
15.2.6	Loss of All Grid Connections
15.2.7	Loss of All Feedwater Flow
15.3.1	Trip of Both Recirculation Pump Motors
15.3.3	Seizure of One Recirculation Pump

The results of these analyses, using the nominal valve setpoints, demonstrate that the maximum vessel pressure will remain below the 1375 psig limit. The safety/relief valves are direct-acting devices and credit is assumed for all of the valves actuating in the analysis. The relief mode (power actuated) of the safety/relief valves provided at Perry is safety-grade and, therefore, half of the total stamped relieving capacity was assumed to be activated via the relief mode. The remainder of the required relieving capacity was assumed to be activated in the safety mode (spring setpoint). This position is consistent with the requirements of Article NB-7540 of Section III of the ASME Boiler and Pressure Vessel Code and we find this acceptable. For the severe transient of main steam isolation valve (MSIV) closure with a high neutron flux scram, the maximum vessel bottom pressure is estimated to be 1295 psig if all of the safety/relief valves operate as previously discussed. The analysis assumed the plant was operating at 105% of rated steam flow (16.16×10^6 lb/hr) and a vessel dome pressure of 1045 psig.

The 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). The applicant has been requested to submit a plant-specific overpressurization analysis using the ODYN for our review and approval. We will report on this analysis in a supplement to this report.

Standard Review Plan Chapter 5.2.2., (Rev-2), Section II-2A(iii) states (for overpressurization analysis) "The reactor scram is initiated by the second safety grade signal from the reactor protection system." The applicant has based the sizing of the safety/relief valves on the initiation of a reactor scram by the high-neutron flux scram which is the second safety grade scram signal from the reactor protection system following MSIV closure. We believe that the qualification and redundancy of reactor protection system equipment coupled with the fact that the reactor vessel pressure is limited to 110 percent of design pressure provides adequate assurance that the reactor vessel integrity will be maintained for the limiting transient event. Accordingly, we find use of the high-neutron flux scram to be acceptable for Perry.

The initial dome pressure assumed is less than the proposed Technical Specification limit. A sensitivity study was performed for a BWR/3 to investigate the effects of increasing the initial reactor pressure relative to the initial value used in the overpressure protection analysis on the peak system pressure. This analysis showed that increasing the initial operating pressure results in an increase in peak system pressure which is less than half the initial pressure increase. For the Perry project, the proposed technical specification limit on the high reactor pressure scram is 1095 psig. Therefore, since the vessel dome pressure used in the overpressurization analysis was 1045 psig, the maximum increase in the initial pressure would be limited to 50 psi, and the maximum peak system pressure increase during the overpressure design transient would be less than 25 psi. Recirculation pump trip has resulted in an increase of 2-6 psi in calculations for other BWRs. These results indicate that considerable margin is available to Perry before reaching the code limit and that GDC 15 will be satisfied even if increased initial dome pressure and recirculation pump trip are considered. However, since the Perry specific overpressure analysis (as well as all other Chapter 15 transient analyses) were performed assuming an initial dome pressure of 1045 pounds per square inch gauge, it is the staff position that the applicants technical specifications should include an operating pressure limit of 1045 psig for the power operation and startup modes, or an analysis be submitted.

Subject to confirmation by the ODYN reanalysis discussed above, we conclude that the pressure relief system in conjunction with the reactor protection system will provide adequate protection against overpressurization of the reactor coolant pressure boundary, is in conformance with the aforementioned Commission's regulations, applicable regulatory guides, and industry standards, and is therefore acceptable.

5.4.6 Reactor Core Isolation Cooling System

The reactor core isolation cooling (RCIC) system is a high-pressure reactor coolant makeup system that will operate independently of alternating current power supply. The system provides sufficient water to the reactor vessel to cool the core and to maintain the reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system is also designed to permit complete plant shutdown under conditions of loss of normal feedwater flow by maintaining the necessary reactor water inventory until the vessel is depressurized to the point where the RHR system can function in the shutdown cooling mode.

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel through the main feedwater line. Fluid removed from the reactor vessel following a shutdown from power operation is normally made up by the feedwater

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

A limited amount of leakage is to be expected from components forming the reactor coolant pressure boundary (RCPB). Means are provided for detecting and identifying this leakage in accordance with the requirements of General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary." Leakage is classified into two types - identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leaktight. Since this leakage is expected, it is considered identified leakage and is monitored and separated from other leakage (unidentified) by directing it to closed systems as identified in the guidelines of Position C.1 of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

Leakage Detection within the Reactor Building

Sources, disposition, and indication of identified leakage are:

A. Within the drywell.

This leakage is collected by the drywell equipment drain sump that discharges to the radwaste system. Sump fill-up and pump-out times are monitored and activate a control room alarm when outside expected limits. Leakage source identification is accomplished by monitoring the drain lines to the sump. These include the

1) upper containment pool seal drain flow,
2) reactor recirculating pump seal drain flow,
3) valve stem leak-off drain line temperatures, and
4) reactor vessel head seal drain line pressure.

Additionally, temperature is monitored in each of the safety/relief valve discharge lines to the suppression pool.

Leakage from 3) and 4) above would be thermally hot and would flash into steam when the pressure is reduced. It cannot be determined how the steam is condensed so that it reaches the sump and is measured for monitoring purposes.

B. Within the reactor building (external to the drywell).

This leakage is collected by the reactor building equipment drain sump that discharges to the radwaste system. Sump fill-up and pump-out times are monitored and activate a control room alarm when outside expected limits. One source of identified leakage in this area of containment is the upper containment pool liner. Flow in the individual drain lines is monitored and high flow-rate is alarmed in the control room. Leakage ~~trom~~ from the reactor water cleanup (RWCU) system is manifested by decreased sump fill-up time.

Sources, disposition, and indication of unidentifed leakage are:

A. Within the drywell.

This leakage is collected by the drywell floor drain sump that discharges to the radwaste system. Sump fill-up and pump-out times are monitored and alarm in the control room when outside expected limits. Other primary detection methods for small unidentified leakage are cooler condensate flow-rate increases and airborne gaseous and particulate radioactivity increases which also alarm in the control room. (The sensitivity of these primary detection methods is 1 gpm within 1 hour.) The monitoring of pressure and temperature of the drywell atmosphere is used to detect gross unidentified leakage.

B. Within the reactor building (external to the drywell).

This leakage is collected by the reactor building floor drain sump that discharges to the radwaste system. Sump fill-up and pump-out times are monitored and alarm in the control room when outside expected limits. Sources of unidentified leakage in this area include the control rod drive hydraulic system and the RWCU system. The arrangement of the RWCU system allows for the installation of a differential flow measurement system. Flow from the reactor is compared with flow back to the reactor, with high differential flow causing an alarm in the control room and initiating isolation.

Leakage Detection Outside of the Reactor Building

The systems for the detection of leakage external to the reactor building do not attempt to distinguish between identified and unidentified leakage. The areas outside the reactor building which are monitored for primary coolant leakage are: equipment areas in the auxiliary building, the main steam tunnel and the turbine building. The process piping for each system to be monitored for leakage is located in compartments or rooms separate from other systems where feasible so that leakage may be detected by area temperature indications. Each leakage detection system will detect leak rates that are less than the established leakage limits.

A. The main steam tunnel, RHR equipment areas, RCIC equipment area, and RWCU equipment area are monitored by dual element thermocouples for sensing high ambient temperature in the areas and high differential temperature between the inlet and outlet ventilation ducts which service the individual areas. Increases in ambient and/or differential temperature will indicate leakage of reactor coolant into the area. These monitors have sensitivities suitable for detection of reactor coolant leakage into the monitored areas. They provide alarm, indication, and recording in the control room.

B. Leakage detection in the turbine building is accomplished by the use of thermocouples for sensing high ambient temperature in the main steam line areas. These monitors also alarm and indicate in the control room.

C. Leakage detection in each ECCS system compartment is accomplished by monitoring increases in floor drain sump level. These monitors also alarm and indicate in the control room.

D. Gross leakage external to the containment (e.g., process line break outside containment) is detected by low reactor water level, high process line flow, high ambient and differential temperature in the piping or equipment areas, high differential flow, and low main condenser vacuum. These monitors provide alarm and indication in the control room and trip the isolation logic to cause closure of appropriate system isolation valves on indication of excessive leakage.

Intersystem Leakage Monitoring

Radiation monitors are used to detect reactor coolant leakage into cooling water systems supplying the RHR heat exchangers and the RWCU system heat exchangers. These monitoring channels are part of the process radiation monitoring system. At least two process radiation monitoring channels monitor for leakage into each common cooling water header downstream of the RHR heat exchangers and the RWCU system non-regenerative heat exchangers. Each channel will alarm on high radiation conditions indicating process leakage into the cooling water.

The leakage monitoring systems are equipped with provisions that permit testing for operability and calibration during plant operation.

The leakage detection system can reasonably be expected to function following seismic events that do not result in plant shutdown. In addition, the airborne particulate radioactivity monitoring system is designed and constructed to seismic-Category-I standards. Thus, the requirements of General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," with respect to the system's capability to perform its safety function following an earthquake are satisfied. However, it is not known if it is connected to a Class 1E power source.

While indicators and alarms for each leakage detection system are provided in the main control room, it is not known what procedures are available for converting various indications to leakage rates.

The technical specifications are not available at this time. Therefore, this section of the regulatory position cannot be evaluated.

The leakage detection systems provided to detect leakage from components of the RCPB furnish reasonable assurance that structural degradation, which may develop in pressure-retaining components of the RCPB and result in coolant leakage in service, will be detected on a timely basis so that corrective actions can be taken before such degradation could become sufficiently severe to jeopardize the safety of the system, or before the leakage could increase to a level beyond the capability of the make-up system to replenish the loss. [We cannot conclude that the systems are in compliance with the guidance found in Regulatory Guides 1.29 and 1.45 and satisfy the requirements of General Design Criteria 2 and 30. Acceptance of the RCPB leakage detection systems is deferred pending resolution of the following items:

(1) The particulate channel of the fission products monitoring subsystem is qualified for SSE. Verify that it receives its power from a Class 1E source.

(2) The drywell equipment drain sump receives hot and cold reactor coolant leakage. Leakage from "hot" sources such as the reactor vessel head flange vent drain and valve packings may flash into steam which must be condensed to reach the sump. Provide assurace that the steam will be condensed for leak detection monitoring purposes.

For leakage from "cold" sources, the floor drain system is employed. Thus, the floor drain system should be tested periodically for blocked lines. Discuss the surveillance program planned to minimize the potential for drain system blockage.

(3) All of the leakage detection systems have readouts and alarms in the control room. Show how the operator will determine the amount of leakage by observing the indications available to him, including the need for unit conversion (e.g., count rate to gpm, etc.,) and how a record of background leakage is maintained. If the monitoring is computerized, discuss the backup procedures that are to be provided assuming failure of the computer.

(4) Discuss the proposed technical specification that limit the conditions for identified and unidentified leakage and that address the availability of the various instrument types to assure adequate coverage at all times.]

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

ATTACHMENT 1

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, NORTH PERRY, OHIO
Perry Nuclear Power Plant, Unit 1
Docket No. STN 50-440

MATERIALS ENGINEERING BRANCH
COMPONENT INTEGRITY SECTION

5.3.1 Reactor Vessel Materials and 5.2.3 Reactor Coolant Pressure Boundary Materials

The staff of EG&G, Idaho National Engineering Laboratory has reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references which are the basis for this evaluation are set forth in paragraph II.3.a of Standard Review Plan (SRP) Section 5.2.3 and paragraphs II.5, II.6 and II.7 (Appendices G and H, 10 CFR Part 50) of SRP Section 5.3.1 in NUREG 0800 Rev. 1 dated July 1981. A discussion of this review follows.

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, testing, and anticipated transient 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 reator pressure boundary. Materials selection, toughness requirements and extent of material testing were reviewed in accordance with the above criteria subject to the rules and requirements of 10 CFR Part 50 Paragraph 50.55a--"Codes and Standards," 10 CFR Part 50 Appendix G--"Fracture Toughness Requirements," and 10 CFR Part 50 Appendix H--"Reactor Vessel Materials Surveillance Program Requirements."

The Perry Unit 1 construction permit was issued on May 3, 1977. Based upon the construction permit date, 10 CFR Part 50 Paragraph 50.55(a) requires that ferritic materials used for the Perry Unit 1 vessel in the reactor coolant pressure boundary be constructed to Section III of the ASME Code no earlier than the Summer 1972 Addenda of the 1971 edition and that ferritic materials used for pressure retaining piping, pump and valve components in the reactor coolant pressure boundary be constructed to Section III of the ASME Code no earlier than the Winter 1972 Addenda of the 1971 edition. Ferritic materials used for fabrication of pressure vessels, piping, pump and valve components that are part of the reactor coolant pressure boundary were constructed to ASME Code Addenda which satisfy these requirements.

(1) Compliance to Appendix G, 10 CFR Part 50

We have evaluated the applicant's FSAR to determine the degree of compliance with fracture toughness requirements of Appendix G, 10 CFR Part 50. Our evaluation indicates that the applicant complied with Appendix G, 10 CFR Part 50, except for Paragraphs III.B.3, III.B.4, and IV.A.3. Our evaluation of each of these areas follows.

Paragraph III.B.3 requires the calibration of temperature instruments and Charpy V-notch impact test machines used in impact testing shall comply with the requirements of paragraph NB-2360 of the ASME Code. To demonstrate compliance with Paragraph III.B.3 of 10 CFR Part 50, the applicant must certify that the calibration schedule for the CVN impact and drop weight machines conforms to NB-2360 of the ASME Code. If such certification is not available, the applicant must indicate the calibration schedule used and provide a basis for granting an exemption to the exact requirements of Paragraph NB-2360 of the ASME Code.

Paragraph III.B.4 requires individuals performing fracture toughness tests be qualified by training and experience and that individuals demonstrate competency to perform tests in accordance with written procedures. The applicant states that the personnel conducting fracture toughness tests were qualified by experience and training that demonstrated competency to perform tests in accordance with required procedures. No records were required to be kept at the time, as the order date of the Perry components predates the requirements of Appendix G. 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. We conclude that an exemption for not performing the tests in accordance with written procedures is justified.

Paragraph IV.A.3 requires, in part, that ferritic materials for piping, pumps, and valves that are part of the reactor coolant pressure boundary meet the fracture toughness requirements of Paragraph NB-2322 of the ASME Code. Table 5.2.5 of the Perry FSAR indicates that ferritic materials were used for reactor coolant piping and for reactor coolant pressure boundary valve disks. The applicant states that testing is performed in accordance with NB-2322, but has not provided any fracture toughness data for ferritic materials used for reactor coolant piping and valves. To demonstrate compliance with Paragraph IV.A.3 the applicant must certify that all ferritic materials in the reactor coolant pressure boundary piping and valves which are in balance of plant and the nuclear steam supply system meet the requirements of NB-2300 of the ASME Code.

(2) Compliance with Appendix H, 10 CFR Part 50

The materials surveillance program at Perry Unit 1 will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region, resulting from exposure to neutron irradiation and the thermal environment. Under the surveillance program of Perry Unit 1, fracture toughness data will be obtained from material specimens that are representative of the limiting base, weld, and heat-affected zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

The fracture toughness properties of reactor vessel beltline materials must be monitored throughout the service life of Perry Unit 1 by a materials surveillance program that meets the requirements of ASTM Standard E 185-73, "Standard Recommended Practice of Surveillance Tests for Nuclear Reactor Vessels" and Appendix H of 10 CFR Part 50.

We have evaluated the applicant's information for degree of compliance to these requirements. Based on our evaluation we conclude that the applicant has met all the requirements of Appendix H, 10 CFR Part 50 with the exception of Paragraph II.B.

Paragraph II.B of Appendix H requires that the surveillance program comply with ASTM E-185-73. ASTM E-185-73 requires the surveillance capsule materials be removed from beltline reactor vessel base metals and weld samples which represent the material that may limit operation of the reactor vessel during its lifetime. The applicant has identified the limiting plate and weld material, but has not furnished the weld wire type and heat identification, flux type, lot identification and chemical composition. The applicant has not identified from which samples the material surveillance specimens were removed, and has not provided sufficient information to define the lead factors and positions of the withdrawal capsules for the material surveillance program. To demonstrate compliance with Paragraph II.B of Appendix H, the applicant must furnish the data detailed above.

(3) Conclusions for Compliance with Appendices G and H, 10 CFR Part 50

Based on our evaluation of compliance with Appendices G and H, 10 CFR Part 50, we conclude that the applicant has not supplied sufficient information to meet all the fracture toughness requirements of Appendix G and surveillance program requirements of Appendix H. The areas in which additional information is required include Paragraphs III.B.3, and IV.A.3, of Appendix G and Paragraph II.B of Appendix H. These items will remain open in our safety evaluation report until the applicant submits the necessary data. We are granting the applicant an exemption for Paragraph III.B.4, Appendix G.

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 significant 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.

Appendix G, "Protection Against Nonductile Failure," Section III of the ASME Boiler and Pressure Vessel Code, will be used, together with the fracture toughness test results required by Appendices G and H, 10 CFR Part 50, to calculate the reactor coolant pressure boundary pressure-temperature limitations for Perry Unit 1.

The fracture toughness tests required by the ASME Code and by Appendix G of 10 CFR Part 50 will provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly prograting fracture can be established for all pressure retaining components of the reactor coolant boundary. The use of Appendix G, Section III of the ASME Code, as a guide in establishing safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations will provide adequate safety margins during operating testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the fracture toughness requirements of General Design Criterion 31.

The materials surveillance program, required by Appendix H, 10 CFR Part 50, will provide information on material properties and the effects of irradiation on material properties so that changes in fracture toughness of material in Perry Unit 1 reactor vessel beltline caused by exposure to neutron radiation can be properly assessed, and adequate safety margins against the possibility of vessel failure can be provided.

5.3.2 Pressure-Temperature Limits

The staff of EG&G, Idaho National Engineering Laboratory has reviewed the applicant's pressure temperature limits for operation of their reactor vessels. The acceptance criteria and list of references which are the basis for this evaluation are set forth in the Standard Review Plan (SRP) Section 5.3.2 of NUREG-0800 Rev. 1 dated July 1981. A discussion of this review follows.

Appendix G, "Fracture Toughness Requirements, and Appendix H, "Reactor Vessel Material 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 boundary at least as great as the safety margins recommended in the ASME Boiler and Pressure Vessel Code Section III, Appendix G, "Protection Against Nonductile Failure." Appendix G of 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 nonductile 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.

The applicant has submitted pressure temperature limits for Perry Unit 1 for normal operation and testing, and they comply with the requirements in Appendix G, 10 CFR Part 50.

5.3.3 Reactor Vessel Integrity

We have reviewed the following FSAR sections related to the reactor vessel integrity for Perry 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.

The staff of EG&G, Idaho National Engineering Laboratory has reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, the pressure temperature limits for operation of the reactor vessels, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references which are the basis for this evaluation are set forth in paragraphs II.2, II.6 and II.7 (Appendices G and H, 10 CFR Part 50) of Standard Review Plan (SRP) Section 5.3.3 in NUREG 0800 Rev. 1 dated July 1981. A discussion of this review follows.

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 the following items:

Paragraph III.B.3, Appendix G: The applicant has not addressed the calibration of temperature instruments and Charpy V-notch impact test machines.

Paragraph III.B.4, Appendix G: The applicant has stated that individuals performing the tests were qualified by experience and training, but that no records were kept at that time. We conclude that although no records were kept, the experience and training of the personnel are sufficient to justify an exemption to Paragraph III.B.4 of Appendix G.

Paragraph IV.A.3, Appendix G: The applicant has not submitted any fracture toughness data on the reactor coolant piping and reactor coolant boundary valve disks.

Paragraph II.B, Appendix H: The applicant has not furnished the weld wire type and heat identification, flux type, lot identification, and chemical composition of the surveillance material. The applicant has not identified from which samples the material surveillance specimens were removed. The applicant has not provided sufficient information to define the lead factors and has not provided a sketch which indicates the azimuthal location for each capsule relative to the reactor core.

Until the applicant has supplied the information necessary to complete our evaluation of compliance with Appendices G and H, 10 CFR Part 50, we cannot complete our evaluation of the structural integrity of the reactor vessels of Perry Unit 1.

analysis was 1045 psig, the maximum increase in the initial pressure would be limited to 50 psi, and the maximum peak system pressure increase during the overpressure design transient would be less than 25 psi. Recirculation pump trip has resulted in an increase of 2-6 psi in calculations for other BWRs. These results indicate that considerable margin is available to Perry before reaching the code limit and that GDC 15 will be satisfied even if increased initial dome pressure and recirculation pump trip are considered. However, since the Perry specific overpressure analysis (as well as all other Chapter 15 transient analyses) were performed assuming an initial dome pressure of 1045 pounds per square inch gauge, it is the staff position that the applicants technical specifications shou[illegible] include an operating pressure limit of 1045 psig for the power operation and [illegible]tup modes, or an analysis be submitted.

Subject to confirmation by the ODYN reanalysis discussed above, we conclude that the pressure relief system in conjunction with the reactor protection system will provide adequate protection against overpressurization of the reactor coolant pressure boundary, is in conformance with the aforementioned Commission's regulations, applicable regulatory guides, and industry standards, and is therefore acceptable.

5.4.6 Reactor Core Isolation Cooling System

The reactor core isolation cooling (RCIC) system is a high-pressure reactor coolant makeup system that will operate independently of alternating current power supply. The system provides sufficient water to the reactor vessel to cool the core and to maintain the reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system is also designed to permit complete plant shutdown under conditions of loss of normal feedwater flow by maintaining the necessary reactor water inventory until the vessel is depressurized to the point where the RHR system can function in the shutdown cooling mode.

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor vessel through the main feedwater line. Fluid removed from the reactor vessel following a shutdown from power operation is normally made up by the feedwater system and supplemented by inleakage from the control rod drive system. If the feedwater system is inoperable, the RCIC system starts automatically when the water level in the reactor vessel reaches the level two (L2) trip setpoint or is started by the operator from the control room. The system is capable of delivering rated flow within 30 seconds of initiation. Primary water supply for the RCIC system comes from the condensate storage tank with a secondary supply from the suppression pool.

The RCIC system was compared to designs and capacities of similar plant systems via comparison tables in Section 1.3 of the final safety analysis report (FSAR), and no unexplained departures from previously reviewed plants were determined.

RCIC design operating parameters are consistent with expected operational modes as noted in Figures 5.4-10 "RCIC Process Diagram" of the FSAR and this complies with the requiremnts of GDC 34 regarding residual heat removal. Essential components of the RCIC system are designated seismic Category I in accordance with Regulatory Guide 1.29 and Quality Group B in accordance with Regulatory Guide 1.26 as discussed in Section 3.2 of this report. Preoperational and initial test program are discussed in Section 14 of this report.

The RCIC system is housed within the auxiliary building which provides protection against wind, tornadoes, floods, and other weather phenomena. Compliance with the requirements of Criterion 2 of the General Design Criteria in this regard is discussed in Section 3.8 of this report. Since the condensate storage tank, which is the normal source of water for this system, is not a seismic Category 1 structure an automatic safety-grade suction switchover to the suppression pool has been provided to ensure a water supply in the event of a safe shutdown earthquake and concomitant failure of the condensate storage tank.

In addition, the system is protected against pipe whip inside and outside containment as required by Criterion 4 of the General Design Criteria, as discussed in Section 3.6 of this report.

The high pressure core spray and reactor core isolation cooling systems are located in different corners of the auxiliary building for additional protection against common mode failures. They use different energy sources for pump motivation (steam turbine for reactor core isolation cooling pump, electric power for high pressure core spray pump) and different power systems for control power. This diversity conforms to the requirements of Section 5.4.6 of the Standard Review Plan.

To protect the RCIC pump from overheating, the reactor core isolation cooling system contains a miniflow line which discharges into the suppression pool when the line to the reactor vessel is isolated. When sufficient flow to the vessel is achieved, a valve in the miniflow line automatically closes thus directing all flow to the reactor.

The reactor core isolation cooling system has its own fill pump to keep the piping full, thereby protecting the system against water hammer. The fill pump operates continuously to keep the system full and pressurized. There is a pressure switch in the discharge of the fill pump to alarm before the pressure drops below the point where draining could occur. The applicant has also agreed to include periodic venting of the high point of the system in plant technical specifications.

A high point vent is provided and the system will be checked at least once every 31 days to assure that the lines are filled. The reactor core isolation cooling system includes a full flow test line with water return to the condensate storage tank for periodic testing. Technical specifications will include a flow test at least every 92 days and a system functional test at least every 18 months with simulated automatic actuation and verification of proper automatic valve position. In both tests verification is obtained that the reactor core isolation cooling pump will develop a minimum flow of 700 gallons per minute.

Isolation between the reactor coolant system and the reactor core isolation cooling system is provided by: (1) two check valves and a closed DC powered valve in the reactor core isolation cooling system discharge line, and (2) two normally open AC powered valves in the steam line to the reactor core isolation cooling steam turbine. We require that the motor operated valves be classified category A and the check valves category A/C in accordance with the provisions of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and that these valves be leak tested periodically. Specific testing requirements for these valves are discussed in Section 3 of this report.

High ambient temperature in the equipment and pipe area will initiate automatic isolation of the system. Spurious ambient temperature signals from in and around the system should not prevent the system from operating when needed. We requested information from the applicant to verify that the high temperature trip setpoints are properly set.

By letter dated October 30, 1981 from Dalwyn R. Davidson (CEI) to R. L. Tedesco (NRC), CEI submitted the additional information. Thermocouples are installed in such a way that they are insensitive to sharp variations in outside air temperature. The trip settings for isolation of the system will be determined by a calculation using the heat balance for the normal operational room environment and introducing a heat release caused by a predetermined steam leakage rate into the area. The resulting temperature transient will be analyzed to obtain the most suitable set point. We find this acceptable.

The suction piping of the RCIC system is designed for low pressure. A relief valve is therefore provided to protect against overpressurization of the line. Rupture discs in the steam turbine exhaust line are used to protect the turbine casing from over-pressurization. To protect the check valve in the turbine exhaust line, a sparger is installed in the line to suppress flow-induced oscillations due to steam bubble collapse in the suppression pool.

Items II.K.3.13 and II.K.3.15 of the Action Plan (NUREG-0737) address restart capability of the RCIC on low water level, the potential for separating the RCIC and HPCS initiation levels, and the prevention of inadvertent RCIC isolation or trip due to spurious signals. The safety evaluation of these items are given in section 22 of this SER.

The reactor core isolation cooling system is capable of supplying coolant to the reactor following feedwater isolation and reactor shutdown under normal and accident conditions. The reactor core isolation cooling system conforms to the requirements of General Design Criterion 5 in that the RCIC System is not shared between units; Criterion 29 (in conjunction with the HPCS System) through quality controlled construction and periodic testing; Criterion 33 (in conjunction with HPCS) in that operation with only offsite or only onsite power is possible for protection against small breaks; Criterion 34 (in conjunction with HPCS) in that residual heat removal while still at high pressure can be accomplished assuming a single failure and with or without offsite power; and Criterion 54 in that suitable leak detection and isolation capability is provided on piping penetrating containment. Review of the drawings, component descriptions, and design criteria for the reactor core isolation cooling system were conducted and, on the basis of this review, we conclude that the design of the reactor core isolation cooling system conforms to the Commission's regulations and to the applicable regulatory guides, and is therefore acceptable.

5.4.7 Residual Heat Removal System

The residual heat removal (RHR) system comprises three independent loops; each loop contains its own motor driven pump, piping, valves, instrumentation and controls. Loops A, B, and C have a suction source from the suppression pool and each is capable of discharging water to the reactor vessel via a separate nozzle or back to the suppression pool via a full flow test line. In addition, the A and the B loops have heat exchangers which are cooled by the emergency service water system. They can also take suction from the reactor recirculation system or fuel pool, and can discharge into the reactor via the feedwater line, or into the fuel pool, or to the containment spray spargers. Loops A and B also have connections to reactor steam via the reactor core isolation cooling (RCIC) steam line and can discharge condensate to the RCIC pump suction or to the suppression pool.

The RHR system is used in conjunction with the main steam and feedwater systems (main condenser), or with the RCIC system in conjunction with the safety/relief valves to cool down the reactor coolant system following shutdown. Interlocks are provided on motor operated valves which interface between the reactor coolant system and the RHR System for protection against overpressurization. In the event the RHR isolation valves are inoperative, an alternative shutdown method is used. In this method, water is pumped from the suppression pool through the RHR heat exchangers and into the reactor vessel. The vessel water is allowed to overflow the steam lines and discharge back to the suppression pool via the discharge lines of those safety relief valves which are part of Automatic Depressurization System.

The RHR system operates in five different modes:

(1) Shutdown cooling,

(2) Steam condensing,

(3) Suppression pool cooling,

(4) Containment spray cooling, and

(5) Low pressure coolant injection.

All five modes of operation use the same hardware. Only the shutdown cooling and the steam condensing modes are covered in this section. Modes (3), (4), and (5) are reviewed through other sections of the Standard Review Plan (6.2 and 6.3).

The normal operational mode of the residual heat removal system is the shutdown cooling mode, which is used to remove decay heat from the reactor core to achieve and maintain a cold shutdown condition. The steam condensing mode is used to condense steam while the reactor is isolated from the main condenser and vessel level is being maintained by the reactor core isolation cooling system. The heat removed in the RHR heat exchangers is transported to the ultimate heat sink by the emergency service water system.

The RHR system was compared to designs and capacities of such systems in similar plants and no unexplained departures from previously reviewed plants were found. The RHR system is designed to operate, with or without offsite power with a single active failure. Control of the RHR system is accomplished from the control room. Using the system process diagrams, P&IDs, system safety analysis, and component performance specifications, it was determined that the system provided at Perry has the capacity to bring the reactor to cold shutdown conditions in a reasonable period of time, assuming operation of only safety-grade equipment.

Isolation between the RHR suction line from the reactor coolant system recirculation loop is provided by an inside containment isolation valve and an outside containment isolation valve. Each valve is interlocked with a separate switch which prohibits opening of the associated valve if the recirculation loop pressure exceeds the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure. An operator error cannot open either valve at a pressure above the shutdown range. Isolation between the RHR shutdown return to the reactor coolant system feedwater line and the RHR connection to the RCIC head spray both have a check valve and globe valve at the pressure boundary. The globe valve has a pressure interlock which prevents opening of the valve due to operator error when the higher pressure system exceeds the shutdown range. This same interlock initiates valve closure on increasing reactor pressure.

We require that the motor-operated valves be classified Category A and the check valves Category A/C in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and that they be tested periodically. Specific testing requirements for these valves are discussed in Section 3 of this report. Relief valves are provided in each of the low-pressure lines that interface with the reactor coolant system. Isolation of the reactor coolant system from the residual heat removal system in this manner complies with the requirements of Section 5.4.7 of the Standard Review Plan and is acceptable. The drywell and suppression pool spray lines have no pressure isolation requirements from the reactor coolant system. The containment isolation requirements of the residual heat removal system are discussed in Section 6.2 of this report.

The residual heat removal system is designed to the seismic Category I requirements of Regulatory Guide 1.29 as discussed in Section 3.2 of this report. It is housed in the auxiliary building for protection against the effects of flooding, tornadoes, hurricanes, and other natural phenomena. Compliance with Criterion 2 of the General Design Criteria in this regard is discussed in Section 3.8 of this report. Compliance with Regulatory Guide 1.26 regarding Quality Group classifications is discussed in Section 3.2 of this report. The containment isolation requirements of Criteria 55, 56, and 57 of the General Design Criteria are discussed in Section 6.2 of this report. Systems used for cooling the residual heat removal system conform to the requirements of Criteria 44, 45, and 46 of the General Design Criteria, as discussed in Section 9.2 of this report. Those portions of the residual heat removal system which are also part of the emergency core cooling system are designed to operate under both normal and accident conditions. The system is protected against missiles (discussed in Section 3.__ of this report) and pipe whip (discussed in Section 3.6 of this report). In this way, the residual heat removal system complies with the requirements of Criterion 4 of the General Design Criteria.

There is only 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 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 provided with emergency power supplies. To assure the long term operability of the automatic depressurization system (ADS) valves for the alternate shutdown cooling mode, two air receivers have been provided to recharge the ADS valve accumulators. The receiver capacity is sufficient to account for system leakage in order to allow a valve to remain open for a period of seven days without replenishment. For longer periods of time the receivers/accumulators can be recharged using compressed air cylinders and the test connection provided outside the containment on the instrument air supply piping. The accumulators, air receivers, associated valves and all the interconnecting air system piping from the isolation valve outside containment to the ADS valves are designed to the requirements of ASME Section III, Class 2 and 3, as applicable, and are

Seismic Category 1. These alternate cooling provisions satisfy the single failure requirements of Criterion 34 of the General Design Criteria. In order to demonstrate valve capability to provide adequate fluid relief for the shutdown cooling mode of the operation, the applicant is participating in the BWR Owners Group test program to test this capability for valves similar to those used at Perry. For the alternate coolirg mode to achieve cold shutdown requires that the safety/relief valves be qualified to pass water as well as steam. Branch technical position RSB-5-1 of the SRP 5.4.7 also requires testing of this alternate method. It is the staff position that the applicant must verify that the safety/relief valves are qualified to pass water (See Item II.D.1 in Chapter 22). In addition, we requested additional information from the applicant to show sufficient flow capacity exist for all piping and valves required in this alternate shutdown path and also to show that the RHR pump head-flow requirements for the worst path resistance can be met. By letter dated November 20, 1981 from Dalwyn R. Davidson (CEI) to R. L. Tedesco (NRC) the applicant provided the requested results of analyses and test data. The test results indicate flow capacities (water) in excess of total valve capacity to pass the required flow rate for alternate shutdown cooling, i.e., only 1 or 2 valve is required to pass the needed flow. This is acceptable to us.

For the low pressure coolant injection mode, flow is diverted to a miniflow line when low flow is sensed in the injection line, with the water discharging to the suppression pool. The miniflow line is designed to prevent pump overheating when the valves in the injection line are closed because the reactor vessel pressure is too high to permit injection. The valves in the miniflow lines are closed automatically when flow in the injection line is sufficient, thus directing all flow to the reactor. Each train of the residual heat removal system is tested during normal plant operation by pumping water from the suppression pool back into the pool. We require that the low-pressure coolant injection mode operability be verified every 31 days; that every 92 days each pump be shown to start from the control room; and every 18 months that a system functional test be performed. This periodic testing is in conformance with the requirements of Criterion 37 of the General Design Criteria. The preoperational test program for the RHR system is discussed in Section 14 of this SER.

The RHR system design has been compared with the functional, isolation, pressure relief, pump protection, and test requirements of Branch Technical Position RSB 5-1 "Design Requirements of the Residual Heat Removal System" and found to comply with the implementation criteria for Perry.

We have reviewed the drawings, component descriptions, and design criteria associated with the residual heat removal system. On the basis of our review, we conclude that the design of the residual heat removal system conforms to the Commission's regulation and to the applicable regulatory guides and is, therefore, acceptable.

DRAFT SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO OPERATION OF

PERRY NUCLEAR POWER PLANT UNIT NO'S. 1 & 2

CLEVELAND ELECTRIC ILLUMINATING COMPANY

DOCKET NO'S. 50-440, 441

*5.4.8 Reactor Water Cleanup System

I. INTRODUCTION

The reactor water cleanup system continuously removes solid and dissolved impurities from the reactor water through filter demineralizers. The system flow path is from the reactor vessel, through two regenerative and two nonregenerative heat exchangers, through two parallel filter demineralizers and returned to the reactor feedwater line. 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 coolant have been established in the

* This evaluation is based upon applicant information through FSAR amendment 3 and by telecon and telex and is subject to confirmatory review of committed FSAR amendments.

Technical Specifications in accordance with the recommendations of Regulatory Guide 1.56, revision 1 (July 1978). The conductivity will be continuously monitored prior to startup, during power operation, hot standby and cold shutdown, to ensure that the limit will not be exceeded. High conductivity will be annunciated in the control room. Surveillance requirements and limiting conditions for operation are specified in the Technical Specifications for Chemistry of Reactor Coolant Systems. The appropriate corrective actions will be taken when the limits of the conductivity, pH, or chloride concentration in the reactor coolant are exceeded.

Spent resins are not regenerable and are sluiced from the filter demineralizer unit to a backwash receiving tank from which they are transferred to the radwaste system for processing and disposal. To prevent resins from entering the reactor recirculation system in the event of complete failure of a filter demineralizer resin support, a strainer is installed on each filter demineralizer unit. Each strainer and filter demineralizer vessel has a control room alarm that is energized by high differential pressure. In the event of low flow or loss of flow in the cleanup system, flow is maintained through each filter demineralizer by its own holding pump.

The suction line of the reactor coolant pressure boundary portion of the reactor water cleanup system contains two motor-operated isolation valves which automatically close in response to signals from the reactor pressure vessel low water level leak detection system and actuation of the Standby Liquid Control System. The outboard isolation valve will automatically close to prevent damage of the filter demineralizer resins if the outlet temperature of the nonregenerative heat exchanger is high. The nonregenerative heat exchanger is sized to maintain the process temperature required for the cleanup demineralizer resin when the cooling capacity of the regenerative heat exchanger is reduced at times when flow is partially bypassed to the main condenser.

II. EVALUATION AND FINDINGS

The system description and piping and instrument diagram was reviewed in accordance with SRP Section 5.4.8. The basis for acceptance in our review has been conformance of the applicant's design of the reactor water cleanup system with the following regulations and Regulatory Guides: (1) the requirements of General Design Criterion 1 by designing, in accordance with the guidelines of Regulatory Guide 1.26, the portion of the reactor water cleanup system extending from the reactor vessel and recirculation loops to the outermost primary containment isolation valves to Quality Group A and by designing in accordance with Position C.2.c. of the Regulatory Guide 1.26, the remainder of the reactor water cleanup system outside the primary containment (excluding the precoating unit) to Quality Group C; (2) the requirements of General Design Criterion 2 by designing in accordance with positions C.1, C.2, C.3, and C.4 of Regulatory Guide 1.29, the portion of the reactor water cleanup system extending from the reactor vessel and recirculation loops to the outermost primary containment isolation valves to seismic Category I; (3) the requirements of General Design Criterion 14 by meeting the positions of Regulatory Guide 1.56 im maintaining reactor water purity and material compatibility to reduce corrosion potential, and thus reducing the probability of reactor coolant pressure boundary failure; and (4) the requirements of General Design Criteria 60 and 61 by designing a system containing radioactivity by confining, venting and collecting drainage from the reactor cleanup system components through closed systems.

On the basis of the above evaluation, we find that, pending confirmatory review of the amended FSAR, the reactor water cleanup system meets the relevant requirements of General Design Criteria 1, 2, 14, 60 and 61 and the appropriate sections of Regulatory Guides 1.26, 1.29, and 1.56 (Revision 1), and, therefore, is acceptable.

~~*6.1.2 Organic Materials~~

~~I. INTRODUCTION~~

~~This evaluation is conducted to verify that protective coatings applied inside containment meet the testing requirements of ANSI 101.2 (1972) and the quality assurance guidelines of Regulatory Guide 1.54. Compliance with these requirements provides assurance that the protective coatings will not fail under DBA conditions and generate significant quantities of solid debris or combustible gas which could complicat the accident conditions.~~

SECTION 6

ENGINEERED SAFETY FEATURES

6.2 Containment Systems Open Areas

The following summarizes the outstanding items identified in the draft SER.

1. Containment Functional Design

   We have not completed our review of the applicant's model and assumptions used to determine the long term containment pressure and temperature transient following onset of a LOCA. We have requested the applicant to provide additional information so that confirmatory analyses can be performed prior to completing our review.

2. Containment External Pressure

   The design basis accident to determine the containment external design pressure was identified as the inadvertent actuation of the containment spray system during normal plant operation. However, for similar types of containments for other plants, e.g., Grand Gulf, the design basis accident was identified as the actuation of the containment spray system following a break in the Reactor Water Cleanup System. We are still reviewing the applicant's analysis to identify the differences between the model and assumptions used for the Perry Plant from those used for other plants.

3. Subcompartments

   The applicant has not provided justification for the assumptions concerning the blowout panels and air seals used in the design of the reactor cleanup rooms and reactor annulus nor provided the analysis that shows the panels will not become missiles as we requested. In addition, we will have to have the applicant submit those experimental data that support the assumptions regarding the blowout panels and air seals. Alternatively, the applicant may elect to propose a testing program designed to demonstrate that the blowout panels and air seals will function as designed.

   Also, the applicant has not provided the pressure loads acting on the structure and components used to establish the design conditions for the reactor annulus region; nor has he shown that the pipe break analyzed for the drywell head region will result in the highest pressure. We will require the applicant to provide this information so that we may complete our review.

4. Steam Bypass of the Suppression Pool

The applicant has not specified the test pressure or intervals at which the periodic reduced pressure test will be performed. We will require the applicant to commit to performing the reduced pressure test at the pressure needed to maintain the water level in the suppression pool slightly above the elevation of the first row of vents and at intervals not exceeding 18 months. We view this as a confirmatory item.

5. Pool Dynamic Loads

We are currently preparing positions on the generic dynamic loads criteria which will be directly applicable to Perry. The results of the staff review of this matter will be reported on at a later date.

6. Secondary Containment

The applicant has not provided sufficient information regarding the model used and assumptions made in determining the secondary containment negative pressure following onset of a LOCA. In addition, we will require the applicant to commit to performing leakage testing of the secondary containment volume to verify the drawdown time assumed in the analysis prior to completing our review.

7. Containment Isolation System

Many of the lines penetrating containment do not conform to the explicit requirements of General Design Criteria 55, 56 or 57. We are currently reviewing the justifications and plan to meet with the applicant to discuss the areas where additional justification is needed. This matter needs to be resolved prior to completion of our review.

8. Containment Purge System

It is our belief that purging/venting should be minimized during normal reactor operation. Therefore, we will require the applicant to provide an estimate of the number of hours per year that purging is expected through each particular valve and to provide justification for these estimates. This matter needs to be resolved prior to completion of our review.

9. Combustible Gas Control

The applicant has not provided the information needed to determine that the amount of hydrogen that would be evolved from the metal-water reaction calculated from a core-wide average depth of 0.23 mils of the fuel cladding is greater than that produced from five times the maximum calculated reactions under 10 CFR Part 50.46. We will require the applicant to provide this information prior to completion of our review.

10. Containment Leakage Testing

We are currently reviewing the applicant's leakage testing program for compliance with the requirements of Appendix J. We plan to meet with the applicant to discuss our results as soon as we complete our review.

~~CONTAINMENT SYSTEMS BRANCH~~
~~REQUEST FOR ADDITIONAL INFORMATION~~
~~PERRY NUCLEAR POWER PLANT UNITS 1 AND 2~~
~~DOCKET NOS 50-440/441~~

Section 6.2- Containment Systems -- OPEN AREAS (QUESTIONS

480.31 The response to question 480.4 was inadequate. Provide the information requested regarding the subcompartment analysis, blowout panels, and the experimental data or test program to support the analysis. Also, provide the same information for the air seals in the reactor annulus region.

480.32 The response to question 480.5 was incomplete. Provide the following additional information:

a) Transient loading on the major components and structure in the reactor annulus region that was used to establish the adequacy of the design. This should include the load forcing functions (e.g., f(t), fx(t), fy(t)), and transient moments (e.g., M(t), Mx(t), My(t), as resolved about a specific, identified coordinate system.

b) Provide the projected area used to calculate these loads and identify the location of the area projection on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluation of the loads and moments can be made.

c) Provide plan and elevation drawings of the biological shield wall annulus region in enough detail to verify the nodalization model used in the analysis.

480.33 The response to question 480.12 is inadequate. Provide a detailed discussion of the reason why the Perry Nuclear Power Plant containment needs to be purged, and an estimate of the number of hours per year that purging is expected through each particular valve. In addition, specify during what modes of operation each purge system will be used.

480.34 The response to question 480.15 states that valves and end caps that are under administrative control are not considered potential leakage paths. It is our position that all test lines between the isolation valves be treated as branch lines. Therefore, include these valves in the containment isolation valve tables and commit to testing them in accordance with Appendix J.

430.35 Provide and justify the model and assumptions made in determining the secondary containment negative pressure following a LOCA. Include in the discussion the time the annulus exhaust gas treatment system is actuated, exhaust flow rate, inleakage assumed, outleakage assumed, coefficient of linear expansion of steel shell, modulus of elasticity of steel shell, time to depressurize to minus 0.25 inch of water, and the time period that direct leakage was assumed in the offsite dose analysis. Also, provide a discussion of the leakage testing program for the secondary containment volume to verify the drawdown time assumed in the analysis.

480.36 In the subcompartment of the drywell head, the double-ended break of the RCIC steam line was determined to result in the rupture that produce the maximum pressure. Justify that the RCIC is the design basis accident for this subcompartment since other high energy lines are in the near proximity of this subcompartment, e.g., the MSL.

480.37 Provide a detailed discussion and justify the model and assumptions used in determining the long term containment pressure and temperature response. The discussion should contain enough information so that confirmatory analyses can be performed.

480.38 It is our position that the periodic reduced pressure test to verify the drywell steam bypass leakage rate must be performed at the pressure needed to maintain the water level in the suppression pool slightly above the elevation of the first row of vents and at intervals not exceeding 18 months. Provide your commitment to comply with this position.

480.39 In regard to the analysis of the hydrogen accumulation in the drywell and containment, provide the following information regarding metal water reaction:

1. Surface area of Zircaoy fuel cladding;
2. Mass of Zircaloy fuel cladding; and
3. Percent of core cladding that was calculated to react with water in demonstrating compliance with Section 50.46(b)(3) of 10 CFR Part 50.

480.40 The response to questions 480.30 regarding a description of your program to improve the hydrogen control capability at Perry 1/2 is deficient. A program description needs to be provided in order for the staff to complete its review in a timely manner.

We need a description of 1) the system you propose to install; 2) the installation schedule; 3) its design bases; and 4) your research programs (including schedules) designed to demonstrate and/or confirm efficacy of the proposed system.

ENCLOSURE ~~- 1~~

~~OPEN~~ ISSUES

~~Overpressurization Protection (5.2.2)~~

~~1. The applicant must submit for our review and approval a plant specific overpressurization analysis using the ODYN code and including the effect of recirculation pump trip.~~

~~2. The applicant's technical specifications should include an initial operating pressure limit of 1045 psig for power operation and startup modes.~~

Emergency Core Cooling Systems (6.3)

3. We require the applicant to provide calculations to verify that flashing would not occur at any point in the ECCS pump suction lines as a result of local elevation changes in the piping run.
4. We require the applicant to verify that the LPCS and LPCI injection valves are interlocked to prevent them from opening unless reactor pressure is below the design pressures of the low pressure systems or to modify the design to conform to one of the acceptable designs given in Section 6.3 of the SRP.
5. We require clarification from the applicant regarding the high drywell pressure interlocks on the HPCS valve injection valve logic.
6. We require a plant specific LOCA analysis for Perry. We require a commitment from the applicant for submittal of the LOCA analysis before fuel loading.
7. We require the applicant to address the inadvertent closure of the FCV in the reactor recirculation system as a single failure in the LOCA analysis.
8. We require additional information to verify that no operator action is required until 20 minutes after the LOCA.
9. We require confirmatory documentation to verify the safety valve position indication method.

~~failure; and (4) the requirements of General Design Criteria 60 and 61 by designing a system containing radioactivity by confining, venting and collecting drainage from the reactor cleanup system components through closed systems.~~

~~On the basis of the above evaluation, we find that, pending confirmatory review of the amended FSAR, the reactor water cleanup system meets the relevant requirements of General Design Criteria 1, 2, 14, 60 and 61 and the appropriate sections of Regulatory Guides 1.26, 1.29, and 1.56 (Revision 1), and, therefore, is acceptable.~~

*6.1.2 Organic Materials

I. INTRODUCTION

This evaluation is conducted to verify that protective coatings applied inside containment meet the testing requirements of ANSI 101.2 (1972) and the quality assurance guidelines of Regulatory Guide 1.54. Compliance with these requirements provides assurance that the protective coatings will not fail under DBA conditions and generate significant quantities of solid debris or combustible gas which could complicat the accident conditions.

In the FSAR the applicant indicates that the coating systems used on exposed surfaces inside the containment have been qualified in accordance with ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," American National Standards Institute (1972). The applicant also stated that the protective coating system for the containments are applied in accordance with Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants."

II. EVALUATION AND FINDINGS

Organic materials inside containment were reviewed in accordance with SRP Section 6.1.2. Based on the applicant's compliance with the applicable Regulatory Guide and ANSI Standard, we find that pending confirmatory review of the amended FSAR, the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on the applicant having met the positions of Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants" and the testing requirements of ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities." These measures demonstrate their suitability to withstand a postulated design basis accident (DBA) environment.

The control of combustible gases that can potentially be generated from the organic materials and from qualified and unqualified paints is reviewed under Section 6.2.5. The consequences of solid debris that can potentially be formed from unqualified paints are reviewed under Section 6.2.2.

~~*9.1.3 Spent Fuel Pool Cleanup System~~

~~I. INTRODUCTION~~

~~The spent fuel pool cleanup system is designed to maintain optical clarity and to remove corrosion products, fission products, and impurities from the spent fuel pool water.~~

~~*This evaluation is based upon applicant information through FSAR amendment 3 and by telecon and telex and is subject to confirmatory review of committed FSAR amendments.~~

~~Draft Safety Evaluation Report~~

~~Containment Systems Branch~~

~~Perry Nuclear Power Plant~~

~~Units 1 and 2~~

~~Docket Nos. 50-440/441~~

6.2 Containment Systems

The containment systems for the Perry Nuclear Power Plant, Units 1 and 2 (Perry 1/2) include a Mark III-type containment structure as the primary containment and a secondary containment surrounding the primary containment. The secondary containment is designed to confine the leakage from the primary containment of airborne radioactive materials.

6.2.1 Containment Functional Design

The containment design for this plant has been given the name Mark III. The Mark III design has three main design features: 1) a drywell surrounding the reactor pressure vessel and a large part of the reactor coolant pressure boundary; 2) a suppression pool that serves as a heat sink during normal operational transients and accident conditions; and 3) a containment structure to prevent the uncontrolled release of radioactivity to the environment. Figure 6.1 shows the principal features of the Mark III containment relative to a Mark I containment.

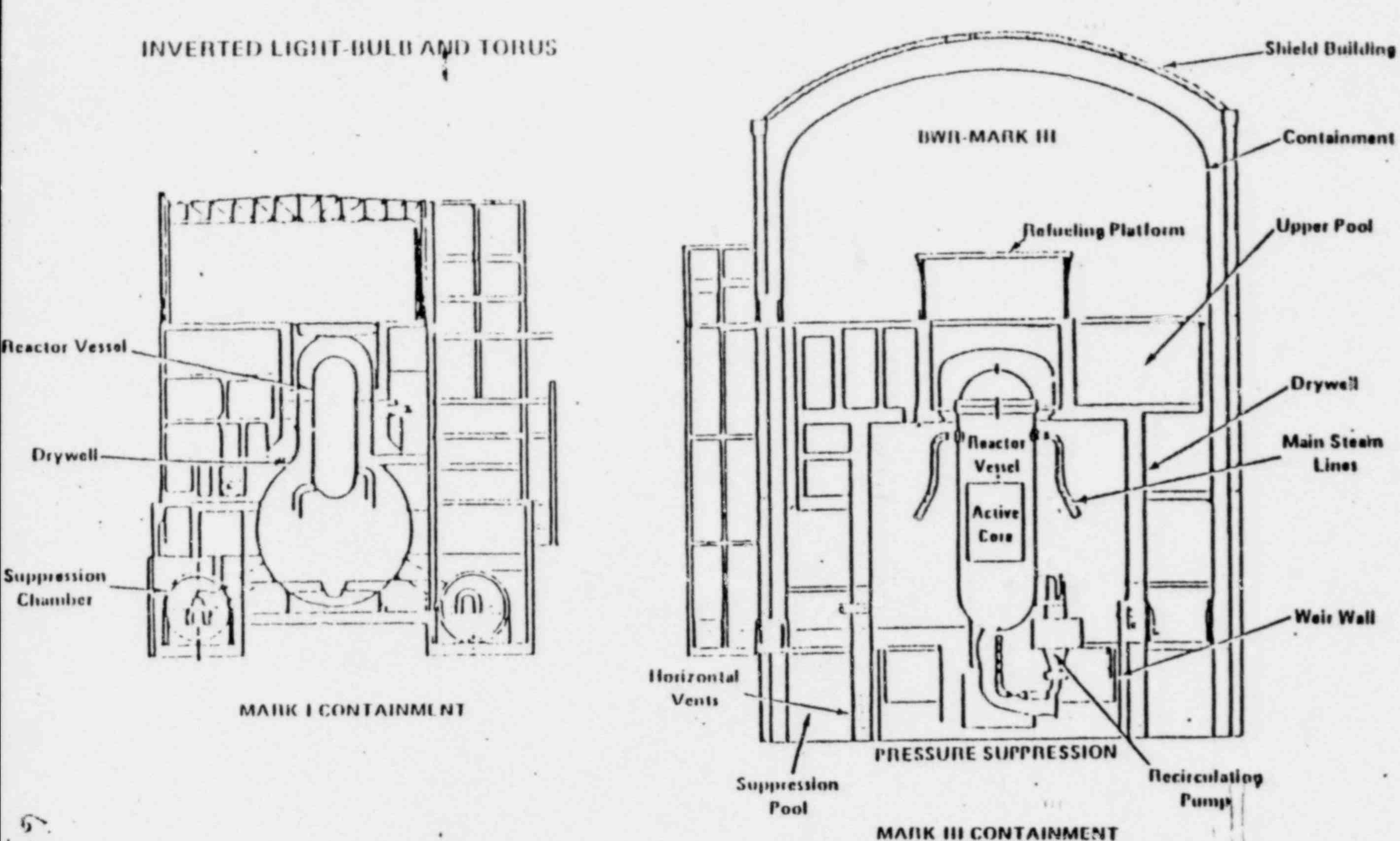


FIGURE 6.1 MARK I AND MARK III CONTAINMENTS

The drywell is designed as the temporary retention boundary which separates the reactor pressure vessel and the recirculation system from the suppression pool and the containment annulus. If the loss-of-coolant accident (LOCA) should occur, the drywell channels released steam around the weir wall and through the horizontal vents and into the suppression pool. Thus, the energy released to the containment annulus is minimized. The drywell has a net free volume of 277,685 cubic feet and is designed for an internal pressure of 30 pounds per square inch gauge (psig) and a temperature of 330 degrees Fahrenheit (°F).

The suppression pool is a 360° annular pool of water located at the bottom of the containment and retained between the containment wall and weir wall, with the drywell wall located between the two. The weir wall is a 360°, reinforced concrete wall located inside the drywell and 30 inches from the drywell wall. At the proposed nominal water height, 18 feet, 3 inches, the volume of water in the suppression pool is 118,625 cubic feet. The suppression pool serves both as a heat sink for postulated transients and accidents and as the source of cooling water for the emergency core cooling systems. In the case of transients that result in a loss of the main heat sink, energy would be transferred to the pool by the discharge piping from the reactor system's safety/relief valves. In the event of a LOCA due to a line break within the drywell, the drywell atmosphere will be vented to the containment via horizontal vents in the drywell wall. Located in the vertical section of the drywell wall and below the suppression pool water level, are 120 vent holes each having a 28 inch diameter. These vent holes are arranged in 40 circumferential columns of three vents. The three vent hole centerlines in each column are located sixteen feet; eleven feet, six inches; and seven feet above the bottom of the suppression pool. In the event of a LOCA, the pressure will rise in the drywell due to the release of reactor coolant, and force the level of water down in the weir annulus. When the water level has been depressed to the level of the first row of vents, the differential pressure will cause air, steam, and entrained water to flow from the drywell into the suppression pool. The steam will be condensed in the pool; and the air driven from the drywell will be transferred into the containment.

The containment is a free standing steel structure consisting of a vertical cylinder, a domed top, and a flat base with a net free volume of 1.14 million cubic feet. The design pressure is 15 psig and the design temperature is 185°F. To satisfy its design basis as a fission product leakage barrier, the containment is designed for a leakage rate of 0.2% of the net free volume per day at the design pressure of 15 psig.

6.2.1.1 Containment Analysis

Our review of the containment included the temperature and pressure responses of the drywell and wetwell to a spectrum of LOCAs; suppression pool dynamic effects during a LOCA and the actuation of one or more reactor coolant system pressure relief valves; the capability to withstand the effects of steam bypass from the drywell directly to the air region of the suppression pool; and the external pressure capability of the containment. In addition, our review considered the applicant's proposed design bases and design criteria for the containment and the analyses and test data in support of the criteria and bases.

Our review included consideration of the loads resulting from pool dynamic-related phenomena. Following a LOCA, an air steam mixture will be forced from the drywell through the vent system into the suppression pool. The air component of the vent flow forms high pressure bubbles in the pool. The air bubbles will create an upward acceleration of the pool surface which can impact internal containment structures. Additional containment loads result as the steam portion of the vent flow condenses in the pool. Actuation of relief valves also results in containment loads. Pressure waves are generated within the suppression pool when the relief valves discharge air and steam into the pool water.

The drywell and containment are divided into subcompartments by internal structures. Our review of the subcompartment designs is discussed in Subsection 6.2.1.6, under "Subcompartment Pressure Analysis."

6.2.1.2 Review of Boiling Water Reactor Containment Technology

There are two basic pressure suppression designs in the United Stated that have preceded the Mark III containment. The Mark I or "lightbulb-torus" and the Mark II or the "over-under design."

A comparison of design parameters for the Mark I, II and III containment types is provided in Table 6.1.

The Mark I containment is the first widely used design for boiling water reactor (BWR) pressure suppression containments. In the Mark I design (see Figure 6.1) the drywell consists of an inverted lightbulb-shaped vessel, and the suppression chamber is a torus shaped steel vessel located below and encircling the drywell. The vent systems consist of ducts, vent header, and downcomers, which enter the suppression pool vertically instead of horizontally like the Mark III vent. The typical design pressure for both drywell and pressure suppression chamber is 56 psig, except for Oyster Creek and Nine Mile Point 1, where the pressure suppression chamber design pressure is 35 psig.

The Mark II containment, like the Mark I and Mark III, js a pressure suppression containment. Figure 6.2 shows the principal features of the Mark II containment. The drywell is in the form of a truncated cone, closed by an elliptical steel dome. The drywell concrete floor serves as a barrier separating the drywell from the suppression chamber. The pressure suppression chamber

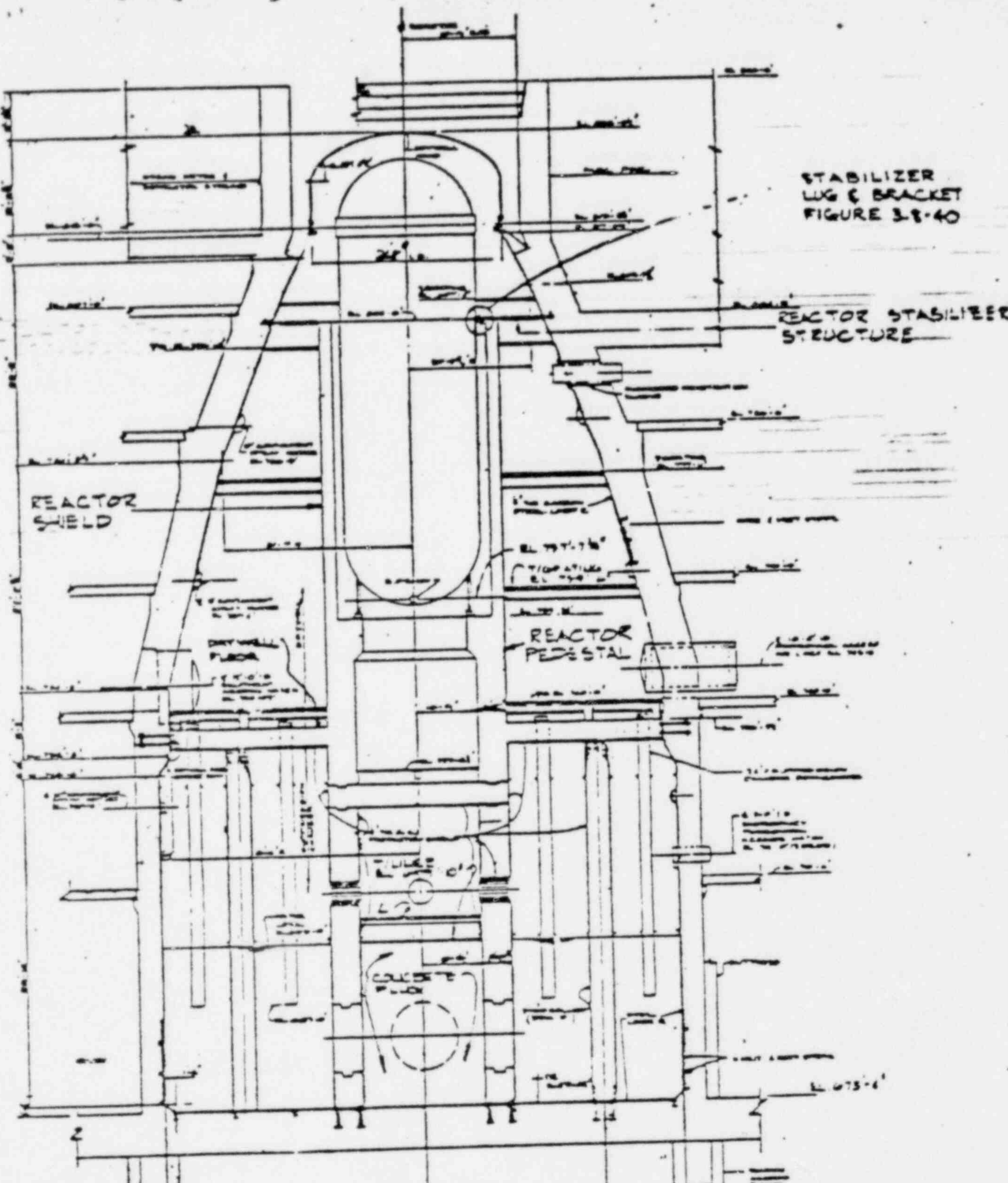


FIGURE 6.2

MARK II CONTAINMENT

Table 6-1

Comparison of BWR Containment Designs

Drywell	Mark I	Mark II	Mark III
	(Hatch 2)	(LaSalle)	(Perry)
type of construction	steel shell	steel-lined post tensioned	reinforced concrete
air volume (cubic feet)	146,266	221,513	277,685
design pressure (pounds per square inch gauge)	56	45	30

Wetwell			
type of construction	steel shell	steel-lined post tensioned	steel-shell
air volume (cubic feet)	109,714	166,400	1.14×10^6
pool volume (cubic feet)	90,550	142,160	118,625
design pressure (pounds per square inch gauge)	56	45	15
leak rate (percent per day)	1.2	0.5	0.2
thermal power (megawatts thermal)	2537	3434	3651
LOCA break area (square feet)	4.378	2.598	3.54
vent area (square feet)	216	295	495
break area/vent area	.0203	.009	.007

is in the shape of a cylinder located below the drywell floor. The vent system consists of straight down pipes (downcomers) which extend through the drywell floor to below the water level in the suppression pool. The typical design pressure for both drywell and pressure suppression chamber is 45 psig.

The Mark III design (see Figure 6.1) is the latest BWR pressure suppression containment design. In this design, the containment (pressure suppression chamber) completely surrounds the drywell. The suppression pool is a 360° annular pool located in the bottom of the containment and retained between the containment wall and the weir wall.

The Mark III type containment proposed for Perry 1/2 is different from the Mark I and Mark II types of containments in three basic ways. First, the BWR/6 type reactor system which is proposed for Perry 1/2 has larger steam lines relative to those of previous BWR core designs. The effect of this difference is that the postulated LOCA involving ruptures in the main steam line results in very nearly equivalent peak drywell pressures when compared with recirculation line ruptures. Therefore, both of these postulated pipe breaks must be considered in determining the design basis accident (DBA) for the Mark III containment pressure response.

Second, the wetwell and drywell of the Mark I and II designs are connected by a vent system which enters the suppression pool vertically at a constant submergence. The Mark III design utilizes a circumferential arrangement of horizontal vents at three different elevations which leads to an additional functional dependence on vent clearing and vent flow phenomena when compared with the Mark I and II types of containment. In addition, because of the relatively large vent areas provided, the peak drywell differential pressure is controlled by vent clearing; i.e., the highest differential pressure across the drywell wall occurs during vent clearing. This places added emphasis on the dynamics of vent clearing, but reduces the impact of vent flow assumptions on drywell pressure. For example, in both the Mark I and II design, the peak drywell pressure occurs in the range of approximately 10 to 50 seconds following onset of the postulated accident, after the vent clearing process and during the vent flow part of the transient. However, for the Mark III design, the peak drywell pressure occurs at about one second, which is during the vent clearing process.

Third, since the volume of the Mark III containment is about four times that of the drywell, the compression of drywell air into the containment during vent flow results in only a small rise in containment pressure. This small effect leads to a long term containment peak pressure which is not specifically related to the size of the reactor coolant system break or to the short-term pressure response.

6.2.1.3 Short-Term Pressure Response

The maximum drywell pressure occurs during the blowdown phase of a LOCA. The maximum containment pressure occurs in the long-term and is discussed in the Subsection "Long Term Pressure Response."

The applicant has performed analyses of various postulated primary system breaks including a double-ended rupture of the recirculation line, a double-ended rupture of the main steam line, an intermediate size rupture of a liquid line, and a small size rupture of the steam line. Results of the analysis indicate that the main steam line break (MSLB) yields the limiting drywell pressure. The applicant, therefore, concludes that the MSLB is the design basis accident for the drywell.

Following the postulated double-ended rupture of a 28 inch MSLB, the mass and energy released from the primary system pressurizes the drywell. As the drywell pressure increases, water in the vent annulus is accelerated downward. At about 0.9 seconds, the first row of vents will be cleared of water and then a mixture of air, steam, and water will flow into the suppression pool where the steam will be condensed and the air will be released to the containment air region.

The water in the vent annulus will continue to accelerate downward resulting in the clearing of the second row of vents at about 1.1 seconds and the third row at about 1.4 seconds. The peak drywell differential pressure occurs at the time the second row of vents is cleared. This is a result of sufficient vent area being uncovered to reverse the pressure transient in the drywell. Due to this phenomenon, the peak pressure is predominantly controlled by the dynamics of vent clearing and only partially influenced by the vent flow assumptions.

The above process is called the vent clearing transient, which occurs less than two seconds following onset of the postulated accident. A detailed discussion of pool dynamics is presented in a later subsection, under "Pool Dynamic Loads."

The containment is pressurized early in the transient by the carryover of noncondensible gases from the drywell. At the end of the blowdown, the drywell pressure stabilizes at a slightly higher pressure than the containment, the difference being equal to the hydrostatic head of vent submergence. The drywell and containment will remain in this equilibrium condition until the emergency core cooling system injection water floods the reactor vessel to the level of the steam line nozzles and the water cascades into the drywell. This results in condensation of the steam in the drywell and a rapid reduction in the drywell pressure. As soon as the drywell pressure drops below the containment pressure, the drywell vacuum breakers will open and noncondensible gases from the containment air volume will flow back into the drywell.

The applicant's calculated peak pressure and temperature for the MSLB are 22.1 psig and 324°F for the drywell and 11.3 psig and 184.6°F for the containment. The design pressure and temperature of the drywell are 30 psig and 330°F, which provide a margin of 36 percent above the peak calculated pressure and 6°F above the peak calculated temperature. The design pressure and temperature of the containment are 15 psig and 185°F, which provides a margin of 33 percent above the peak calculated pressure and essentially no margin for the peak calculated temperature. We performed an analysis of the drywell pressure response using the CONTEMPT-LT 28 computer code. Our calculation of the peak pressure and temperature confirm those calculated by the applicant. Therefore, on the basis of our confirmatory analyses, we conclude that the drywell design pressure of 30 psig and design temperature of 330°F are acceptable.

6.2.1.4 Long-Term Pressure Response

Following the short-term blowdown phase of the accident, the suppression pool temperature and containment pressure will continuously increase due to the input of decay heat and sensible heat into the containment. At the end of the blowdown phase of the accident, the drywell pressure has stabilized at a slightly higher pressure than the containment. This differential pressure corresponds to the hydrostatic head of the submergence of the first row of vents. At a later time, the drywell and containment chamber pressures will equalize due to the return of air from the containment.

During this time period, the ECCS pumps, taking suction from the suppression pool will have reflooded the reactor pressure vessel up to the level of the main steam nozzles. Subsequently, ECCS water will flow out of the break and fill the drywell up to the top of the weir wall, establishing a recirculation flow path for the ECCS coolant. This relatively cold ECCS water will condense the steam in the drywell and bring the drywell pressure down rapidly, at about 309 seconds after onset of the accident. At ten minutes following onset of the accident, the containment cooling mode of the residual heat removal system is activated and supression pool water is circulated through the residual heat removal system heat exchangers, establishing an energy transfer path to the service water system and the ultimate heat sink.

In the long-term analysis, the applicant accounted for potential post-accident energy sources. These include decay heat, sensible heat and metal-water reaction energy. The applicant's long-term model also assumes that the containment atmosphere is saturated and equal to the suppression pool temperature at any time. Therefore, the containment pressure is equal to the sum of the partial pressure of air and the saturation pressure of water corresponding to the pool temperature.

Based on the above assumption, the applicant calculated a peak supression pool temperature of 184.6°F for the most limiting residual heat removal system cooling mode; i.e., only one operating residual heat removal system cooling loop with only one residual heat removal system pump being available. The calculated long term secondary peak containment pressure is about 11.3 psig. The containment is designed for 15 psig and 185°F. The applicant's model and assumptions used for the long term pressure and temperature analysis is still under review by the staff. We have requested additional information from the applicant and will report on this matter in a supplemental report.

6.2.1.5 Containment External Pressure

The transients which could result in significant negative pressure within the containment involve the inadvertent actuation of the containment spray while the containment atmosphere is at high temperature and humidity. The applicant has analyzed two such transients: one transient is the actuation of the containment spray following a break in the Reactor Water Cleanup System; and the other is inadvertent actuation of the containment spray during normal plant operation. The applicant has calculated a negative overall containment differential pressure of 0.72 psig for the latter case. The staff is still reviewing the applicant's analysis.

6.2.1.6 Subcompartment Pressure Analyses

Within both the drywell and containment, internal structures form subcompartments or restricted volumes which are subjected to differential pressures subsequent to postulated pipe ruptures. In the drywell, there are two such volumes: the Reactor Pressure Vessel Annulus which is the annular region formed by the reactor vessel and the biological shield; and the Drywell Head which is a cavity surrounding the reactor pressure vessel head. In the containment, there are four compartments which contain various components of the Reactor Water Cleanup (RWCU) System. There is also a Main Steam tunnel which is located between the drywell and containment.

The applicant has performed pressure response analyses for various postulated pipe breaks within these subcompartments. The blowdown mass and energy release are based on Moody's flow models and full, double-ended pipe ruptures. The analyses of the pressure transients within these subcompartments were performed using the computer code, COMPARE, except for the reactor pressure vessel annulus which used the RELAP 4/MOD 3 computer code.

We have also analyzed the subcompartments pressure response for the various pipe breaks using the computer code COMPARE. In all compartments, our predicted maximum pressure differential was below the design pressure of the subcompartment. However, the applicant has not provided justification for the assumption concerning the blowout panels and air seals used in the design of the reactor water cleanup room and reactor annulus region. We will require the applicant to provide the experimental data that support the assumptions made regarding the blowout panel and air seals or propose a testing program that will demonstrate the blowout panel and air seals will function as designed. Also, the applicant will be required to provide an analysis that shows there will be no missiles generated.

In addition, the applicant has not provided the pressure loads acting on the compartment structure and major components used to establish the support design, nor justified that the pipe breaks analyzed in the drywell head region will result in the highest pressure. We will require the applicant to provide this information. We will report on these matters in a supplemental report.

6.2.1.7 Steam Bypass of the Suppression Pool

During a postulated primary system line break inside the drywell, possible bypass leakage paths between the drywell and containment air space could result in excessive containment pressures. The control of such bypass paths is important to ensure that the design pressure of the containment is not exceeded. There are several potential sources of steam bypass to the suppression pool air space in Perry 1/2. They include cracking of the drywell concrete structure, the drywell vacuum breakers, and penetrations through the drywell structures.

The drywell leakage capacity has been evaluated for a spectrum of primary system rupture areas. The basic model included passive heat sinks and automatic containment spray actuation. The applicant has shown that maximum allowable leakage area of $A/\sqrt{k}$ = 1.68 square foot could exist between the drywell and containment following the limiting line break accident without exceeding the design pressure. We have not completed our review of the applicant's analysis. We will report on this matter in a supplemental report.

In order to assure that the existing steam bypass leakage area is well below the assumed area in the analysis, the applicant has proposed to perform preoperational and periodic leakage tests. The preoperational test will be a high pressure test performed at the drywell design pressure. The periodic test will be performed at a reduced pressure and at intervals not yet specified by the applicant. We will require the applicant to perform the reduced pressure test at the pressure needed to maintain the water level in the suppresion pool slightly above the elevation of the first row of vents and at intervals not exceeding 18 months. We will report on this matter in a supplemental report.

The applicant has proposed to limit the measured leakage to less than ten percent of the allowable bypass leakage. We have concluded that the applicant's acceptance criterion for determining the bypass capability of the drywell is acceptable.

6.2.1.8 Pool Dynamic Loads

Several phenomena have been identified in our review of the Mark III containment that could result in dynamic loading of structures located in and above the suppression pool. They are related to (1) pool response to the LOCA; and (2) pool response due to relief valve operation, generally associated with plant transient conditions. These phenomena are describd in more detail below.

6.2.1.8.1 LOCA Pool dynamics

Following a LOCA in the drywell, the drywell atmosphere will be compressed due to blowdown mass and energy addition to the volume. Following vent clearing an air/steam/water mixture will be forced from the drywell through the vent system and injected into the suppression pool, approxiamtely 7-10 feet below the surface. The steam component of the flow mixture will condense in the pool, while the air will be released in the pool as high pressure bubbles. The continued addition and expansion of air causes the pool volume to swell resulting in an acceleration of the surface vertically upward. Due to the effect of buoyancy, air bubbles will rise faster than the pool water mass and will eventually break through the swollen surface and relieve the driving force behind the pool. Due to the dynamics of vent clearing and vent flow and the vertical motion of the pool water mass, structures forming the suppression pool boundary, structures located within the pool, and structures located above the pool could be subject to hydrodyamic loads.

6.2.1.8.2 Relief Valve Dynamics

Pressure waves are generated within the suppression pool when, on first opening, relief valves discharge high pressure air and steam into the pool water. This phenomenon is referred to as relief valve vent clearing loads which are imparted to pool retaining structures and structures located within the pool. These same structures can also be subject to loads which accompany extended relief valve discharge into the pool if the pool water is at an elevated temperature. This effect is known as steam quenching vibrations.

The Mark III LOCA-related pool dynamic loads were reviewed at the Construction Permit (CP) stage for Perry 1/2 and at the PDA stage for GESSAR-238NI. The staff concluded at the time that the information available was sufficient to adequately define the pool dynamic loads for nuclear plants applying for CPs. Since the issuance of the GESSAR-238NI SER (December 1975), GE has conducted further tests and analyses to confirm and refine the original load definitions.

The staff is currently reviewing this information to arrive at a finalized hydrodynamic load definition that can be utilized by Mark III containment applicants for an OL. The LOCA pool dynamics are being reviewed under Task Action Plan (TAP) B-10, "Behavior of BWR Mark III Containment" and TAP A-39, "Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment." The end product of these two generic programs will be applicable to Perry 1/2.

The results of the staff review of the pool dynamic loads for Perry 1/2 will be reported in a supplement to the SER.

6.2.1.9 Secondary Containment

The secondary containment system includes the structures and systems to be used to control and treat radioactive leakage from the primary containment in the event of a LOCA. For Perry 1/2, the secondary containment structures consist of the shield building. The shield building is a cylindrical reinforced concrete structure which completely surrounds the containment.

The standby gas treatment system (SGTS) maintains the secondary containment at a pressure of -0.25 inches of water and provides cleanup of the potentially contaminated secondary containment volume following a LOCA.

The SGTS is designed to seismic Category I criteria, safety quality group B and consists of redundant recirculation fans and filtration trains each consisting of a demister, electric heater, prefilter, high-efficiency particulate air filter and carbon adsorbers.

Following a postuated LOCA, the pressure in the secondary containment could increase due to inleakage, expansion of the containment steel shell and the starting time required for the standby gas treatment system The applicant has not provided enough information on the analysis of the secondary containment pressure transient to determine if the above phenomenon was considered in the analyses. We have requested the applicant to provide this information.

Operation of only one of the two redundant SGTS train was assumed for the analyses . However, the applicant has not provided the time period that bypass of the secondary containment was assumed to occur for calculating the offsite radiological consequences associated with the pressure transient. We will require the applicant to provide the information.

We will also require the applicant to commit to leakage testing of the secondary containment volumes to verify the drawdown time for reestablishing the -0.25 inches of water gauge pressure. We will report on the resolution this matter in a supplemental report.

Although the primary containment is enclosed by the secondary containment, there are systems which penetrate both the primary and secondary containment boundaries creating potential paths, in the event of a LOCA, through which radioactivity in the primary containent could bypass the leakage collection and filtration systems associated with the secondary containment. A number of these lines contain physical barriers or design provisions which can effectively eliminate leakage, e.g., water seals, closed seismic Category I piping system, or vent return lines to controlled regions. The criteria by which potential bypass leakage paths are determined has been set forth in Branch Technical Position (BTP) CSB-6-3, "Determiniation of Bypass Leakage Paths in Dual Containment Plants." However, the staff has not completed its review of all the potential bypass leakage paths. Also, we will require the applicant to propose a testing program for measuring the fraction of primary containment leakage that may bypass the secondary containment. We will report on the resolution of these matters in a supplemental report.

6.2.2 Containment Heat Removal System

The containment heat removal system includes the piping, valves and mechanical components which will be used to remove energy from the containment to limit temperature and pressure in the drywell and containment following a postulated LOCA. It is an integral part of the residual heat removal system.

The residual heat removal system consists of two complete loops including heat exchangers and main system pumps. Each loop is designed such that a failure in one loop cannot cause a failure of another. In addition, each of the loops and associated equipment is located in a separate protected area of the reactor building to minimize the potential for single failures including loss of onsite or offsite power causing the loss of function of the entire system. The system equipment piping and support structures are designed to seismic Category I criteria.

Provisions have been made in the residual heat removal system to permit inservice of system components and functional testing of active components.

Operating in the containment cooling mode, the residual heat removal pumps take suction from the suppression pool. Flow is then directed through the residual heat removal heat exchangers to the suppression pool, the reactor vessel, or the containment spray headers. The location of system and return lines in the suppression pool facilitates mixing of the return water with the total pool inventory before the return water becomes available to the suction lines. Strainers are provided on the suction line inlets. The applicant has provided analyses of the long-term post-accident containment pressure and temperature response assuming various combinations of containment cooling availability. Our evaluation of this analysis is discussed in Subsection 6.2.1.4.

The applicant analyzed the net positive suction head that is available at the residual heat pump inlets assuming the containment will be at atmospheric pressure and the pool is at saturation temperature. In addition, the applicant designed the suction piping from the suppression pool so that if any one suction strainer is 50 percent plugged, the maximum required net positive suction head to the residual heat removal system pumps during containment cooling will be provided. The above assumptions are in agreement with the provisions of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Sumps," and therefore are acceptable.

The potential for debris to clog the residual heat removal system suction lines was evaluated. Each residual heat removal system pump is provided its own suction line and strainer assembly. The pipe insulation used in the drywell, metal reflective insulation, is of a type to minimize the potential for its breaking away from piping and being carried through the vent system into the suppression pool. This design minimizes the potential of clogging the suction line. Therefore, we find the design of the pump suction structure strainers acceptable.

We conclude pending confirmatory analysis of the long term pressure and temperature analysis that the containment heat removal system can be operated in such a manner as to provide adequate cooling to the containment following a LOCA and will conform to General Design Criteria 38, 39 and 40 and is acceptable.

6.2.3 Containment Isolation System

The containment isolation system includes the containment isolation valves and associated piping and penetrations necessary to isolate the primary containment in the event of a

LOCA. Our review of this system included the number and location of isolation valves, the valve actuation signals and valve control features, the positions of the valves under various plant conditions, the protection afforded isolation valves from missiles and pipe whip, and the environmental design conditions specified in the design of components.

The design objectives of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary to prevent or limit the escape of fission products from a postulated LOCA. The applicant specified design bases and design criteria as well as the isolation valve arrangement to be used for isolation of primary containment penetrations.

The containment isolation system is designed to automatically isolate the containment atmosphere from the outside environment under accident conditions. Double barrier protection, in the form of two isolation valves in series or a closed system and isolation valve, are provided to assure that no single active failure will result in the loss of containment integrity. The containment isolation system components, including valves, controls, piping and penetrations, are protected from internally or externally generated missiles, water jets and pipe whip.

The basis for our acceptance has been the conformance of the containment isolation provisions to the Commission's regulations as set forth in the GDC, and to applicable regulatory guides, our technical positions, the Standard Review Plan and industry codes and standards.

The containment isolation systems are designed to the American Society of Mechanical Engineers Code, Section III, Class 1 or 2 and are classified as seismic Category I design systems.

The containment isolation provisions for the lines penetrating containment conform to the requirements of GDC 55, 56 or 57, as appropriate. As provided by GDC 55 and 56, there are containment penetrations whose isolation provisions do not have to satisfy the explicit requirements of the GDC but can be accepted on some other defined basis. However, the applicant has not justified all the design deviations from the explicit requirements of the GDC. We will require the applicant to provide this justification and report the resolution on this matter in a supplemental report.

6.2.3.1 Containment Purge System

The design of the containment purge and ventilation system consists of both 42-inch and 18-inch lines. The 18-inch line will be used to continuously purge a small amount of the containment atmosphere during normal operation. The applicant has not specified for what modes of operation the 42-inch and 18-inch lines will be used. We will require the applicant to provide this information.

It is our belief that purging/venting should be minimized during reactor operation because the plant is inherently safer with closed purge valves than with open valves requiring valve action to provide containment isolation. Therefore,. we will require a detailed justification for the need to purge, and an estimate of the number of hours per year that purging is expected through each particular valve. We will report on the resolution of these matters in a supplmental report.

6.2.5 Combustible Gas Control

The combustible gas control systems include the piping valves, components and instrumentation necessary to detect the presence of combustible gases within the primary containment and to control the concentration of these gases.

The scope of review of the design and functional capability of the combustible gas control system for Perry 1/2 included drawings and descriptive information of the equipment to mix the containment atmosphere, monitor combustible gas concentration, and reduce combustible gas concentrations within the containment following the design basis accident. The review also included the applicant's proposed design bases for the combustible gas control systems, and the analyses of the functional capability of the system provided to support the adequacy of the design bases.

The bases for our acceptance are the conformance of system design and design bases to the Commission's regulations as set forth in the GDC, and to applicable regulatory guides, branch technical positions, and industry codes and standards.

Following a design basis LOCA, hydrogen may accumulate within the containment as a result of metal-water reaction between the fuel cladding, corrosion of construction materials, and as a result of radiolytic decomposition of the post accident emergency cooling water. The applicant analyzed the production and accumulation of hydrogen from the above sources. The guideline regarding the metal-water reaction states that hydrogen production is five times the maximum calculated reaction under 10 CFR Part 50.46, or that amount that would be evolved from a core wide average depth of reaction into the original cladding of 0.23 mils, whichever is greater, in two minutes. The applicant's analysis of the combustible gas control system was based on the amount of hydrogen that would be evolved from a core wide reaction of an average depth of 0.23 mils of the fuel cladding. However, the applicant has not provided enough information to determine that the hydrogen calculated by this method is greater than that calculated assuming five times the maximum metal-water reaction calculated under 10 CFR Part 50.46. We will require the applicant to provide this information.

The applicant's combustible gas control system consists of three subsystems which are: (1) a drywell purge system; (2) a hydrogen control system; and (3) a backup purge system. The major components of the drywell purge system are compressors, valves, piping and the required instrumentation. The drywell purge system will pump the hydrogen produced within the drywell into the containment. The system consists of redundant components so that any single active failure will not prevent purging of the drywell volume.

The hydrogen control system consists of two 100 percent capability hydrogen recombiners manufactured by Westinghouse. Each unit has a design flow rate of 100 SCFM and located inside the containment. The backup purge system is also provided in accordance with Regulatory Guide 1.7. This system will purge the containment atmosphere, if necessary, through the containment filtration system's charcoal filter trains to reduce the activity released.

The applicant performed calculations of the containment hydrogen concentration in the drywell and in the containment. The applicant calculated a maximum drywell concentration of about 3.5 volume percent occurring at approximately 12 days after the postulated accident. The maximum calculated containment hydrogen concentration is also about 3.5 volume percent. We performed confirmatory calculations using the COGAP-2 computer code. The results of this analysis indicate that the hydrogen concentration will reach 3.5 volume percent at an earlier time than calculated by the applicant, but the combustible gas control system is still capable of performing its design fraction; that is, limit the hydrogen concentraion to less than 4.0 volume percent. We conclude that the design of the combustible gas control system conforms to all applicable regulations, guides, our positions, and industry standards and is acceptable pending verification that the applicant's hydrogen source term for the metal-water reaction is acceptable.

6.2.6 Containment Leakage Testing

The applicant has provided information to demonstrate compliance with the testing requirements of Appendix J to Title 10 of the Code of Federal Regulations, Part 50. Our review of the containment leakage testing program is not complete. We will report on this matter in a supplement to the SER.

6.2.7 TMI-2 Requirements

II.E.4.1 Dedicated Hydrogen Penetrations

II.E.1 Attachment 4, Containment Pressure Monitor

II.F.1 Attachment 5, Containment Water Level Monitor

II.F.1 Attachment 6, Containment Hydrogen Monitor

II.B.8 Rulemaking Proceeding on Degraded Core Accidents.

The applicant has not addressed the TMI action items.

6.3 Emergency Core Cooling System

The emergency core cooling system (ECCS) is designed to provide water to the reactor coolant system in the event of a break in the pressure boundary. The ECCS capability extends to failures as large as a double-ended rupture of the largest pipe carrying water or steam, and spurious safety/relief valve operation.

The basis for the design of the ECCS is to limit damage to the fuel cladding in accordance with Title 10, Code of Federal Regulations, Part 50.46 (10 CFR 50.46). The system must be capable of performing its design function with or without offsite power and with a single failure, including loss of an emergency diesel generator.

6.3.1 System Description

The ECCS consists of the following systems:

(1) High Pressure Core Spray System (HPCS)

(2) Automatic Depressurization System (ADS)

(3) Low Pressure Core Spray System (LPCS)

(4) Low Pressure Coolant Injection System (LPCI)

The HPCS is provided to maintain the reactor vessel water level above the top of the active core in the event of small pipe breaks and to provide spray cooling in case the core is uncovered. Activation of the HPCS does not require the depressurization of the reactor vessel. The system includes a single motor-driven centrifugal pump which takes suction from the condensate storage tank or the primary containment suppression pool. An automatic switching feature is provided. HPCS flow is dependent on the pressure differential that exists between the reactor system and the suction source. Pump characteristic curves indicate that the rated HPCS flow (7200 gpm) is attained at approximately 217 psid. HPCS discharges water into the reactor via a spray sparger mounted on the reactor vessel internal wall above the core. The HPCS system is designed to operate from normal offsite auxiliary AC power or from its own diesel-generator.

The ADS is provided to depressurize the reactor coolant system in the event a small pipe break occurs and the HPCS system cannot maintain reactor vessel water level or fails to start. The ADS employs 8 of the 19 safety/relief valves to reduce system pressure so that the low pressure systems may inject water to cool the core.

The LPCS system is provided to replace reactor vessel water inventory and to supply spray cooling following large pipe breaks in which the core may uncover. The system includes a motor-driven centrifugal pump which takes suction from the suppression pool and discharges water to the reactor vessel via a spray sparger mounted on the reactor vessel internal wall above the core. Pump characteristic curves indicate that rated LPCS flow (8000 gpm) is attained at a pressure of approximately 190 psid. The LPCS sparger is separate from the HPCS sparger. The LPCS system is designed to operate from normal auxiliary AC power or from the standby AC power system. The LPCS pump and RHR pump A are on the same electrical bus and are supplied emergency power by the same diesel generator.

The LPCI system is provided to replace reactor vessel water inventory following large pipe breaks. The system is an operating mode of the Residual Heat Removal (RHR) system which consists of three independent loops (A, B, and C). Each loop has a motor-driven pump (7100 gpm) which takes suction from the suppression pool and supplies water to the reactor vessel via a separate nozzle through the reactor vessel wall. In addition, loops A and B can also take suction from the reactor recirculation system suction or fuel pool, and can discharge into the reactor via a feedwater line, fuel pool cooling discharge, or to the containment spray spargers. RHR loops A and B have heat exchangers which are cooled by the emergency service water system and are used to transfer the decay heat from the reactor core to the ultimate heat sink. The three LPCI (RHR) pumps are powered from AC power buses having standby power source backup supplies. RHR pumps B and C are on the sa electrical bus and receive emergency power from the same diesel generator. RHR pump A is on the same electrical bus as the LPCS pump.

6.3.2 Evaluation

6.3.2.1 Single Failures

We reviewed the system description and piping and instrumentation drawings to assure that abundant core cooling will be provided during the injection phase with and without offsite power and assuming a single failure as required by Criterion 35 of the General Design Criteria. A low reactor vessel water level and/or high containment pressure signal is required to start pumps and open discharge valves.

The applicants provided in Section 6.3.3 of the Final Safety Analysis Report an analysis to demonstrate that the most limiting break size, break location, and single failure had been considered for Perry. The most limiting combinations are tabulated below.

Break Size	Break Location	Single Failure	Systems Remaining
Small	Recirculation Suction Line	HPCS	ADS, LPCS and all LPCI
Intermediate and Large	Recirculation Suction Line	LPCI Diesel Generator	HPCS, ADS, LPCS and one LPCI

The applicant has analyzed main steam breaks inside and outside containment, HPCS line breaks and feedwater line break locations. The analyses have shown these break locations, assuming the worst single failure, are not limiting. These breaks all occur at higher elevations which result in faster depressurization and earlier actuation of the emergency core cooling system. In addition, the reduced loss of inventory results in lower peak clad temperature.

6.3.2.2 Qualification of the Emergency Core Cooling System

The emergency core cooling system is designed to meet seismic Category I requirements in compliance with Regulatory Guide 1.29, "Seismic Design Classification," as discussed in Section 3.2 of this report. It is housed in structures designed for seismic events, tornadoes, floods, and other phenomena in accordance with the requirements of Criterion 2 of the General Design Criteria as discussed in Section 3.8 of this report. Emergency core cooling system equipment is designed in compliance with Regulatory Guide 1.26, "Quality Group Classification and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants," as discussed in Section 3.2 of this report.

Protection of the emergency core cooling system against pipe whip and against discharging fluids in compliance with the requirements of Criterion 4 of the General Design Criteria and Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," is discussed in Section 3.6 of this report. Evaluation of the instrumentation and controls for the emergency core cooling system is discussed in Section 7.3 of the report. Compliance with the inservice inspection requirements of Criterion 36 of the General Design Criteria is discussed in Section 6.6 of this report. Environmental qualification of the emergency core cooling system equipment for operation under normal and accident conditions as required by Criterion 4 of the General Design Criterion is discussed in Section 3.11 of this report.

6.3.2.3 Functional Design

The available net positive suction head for the pumps in the emergency core cooling system should have adequate margin to prevent cavitation and assure pump operability in accordance with Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." We requested additional information from the applicant to verify adequate NPSH for the ECCS pumps. We also requested that the applicant provide calculations verifying that flashing would not occur at any point in the ECCS suction lines as a result of local elevation changes in the piping runs. We will report on these items in a supplement to this SER.

The HPCS incorporates relief valves to protect the components and piping from inadvertant overpressure conditions resulting from either thermal expansion or backpressure leakage into the low pressure portions of the system. The LPCS and LPCI systems are not designed to withstand normal reactor operating pressure. Relief valves are provided in these lines to protect against leakage from the reactor coolant system. Each of the low pressure lines that interface with the reactor coolant system has a testable check valve inside primary containment backed up by a normally closed motor-operated gate valve outside of containment. We requested the applicant to verify the interlock of this valve so that it does not open until the reactor coolant pressure is below the system design pressure. We will report on this item in a supplement to this report.

Containment isolation in accordance with the requirements of Criterion 55 of the General Design Criteria is discussed in Section 6.2 of this report. The periodic testing and leak rate criteria for these valves that isolate the reactor coolant system from the emergency core cooling system are discussed in Section 3.9.6 of this report. The detection of leaks from those portions of the emergency core cooling system within primary containment is discussed in Sections 5.2.5 and 9.3.3. Leak detection for portions of the emergency core cooling system outside of primary containment is provided primarily by the reactor building floor and equipment drain sump leak detection system and by the equipment area high temperature detection system.

All the emergency core cooling systems have miniflow lines to permit a limited amount of flow in the event an isolation valve between the reactor coolant system and emergency core cooling system is closed, for any reason, in order to protect the pumps from overheating. When sufficient flow passes through the injection lines, valves in the miniflow lines automatically shut, diverting all flow to the pressure vessel. The lines from the suppression pool to the suctions of the low pressure coolant injection and low pressure core spray pumps each have an open motor-operated valve outside of containment with controls arranged so that a key is required to unlock a lever to close the valve. The suction of the high pressure core spray from the suppression pool contains a closed motor-operated valve outside the containment designed to open so that the system automatically pumps water from the suppression pool instead of the condensate storage tank when the condensate storage tank water is exhausted.

As a backup to the high pressure core spray system, the automatic depressurization system can be used to depressurize the system and allow the functioning of the low pressure cooling systems in the event of a small break. The air supply to the automatic depressurization system valves is provided in accident conditions by seismically qualified accumulators and receivers to compensate for leakage past accumulator check valves.

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 system pump discharge lines full, the lag time between the signal fo. 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 Perry a fill system consisting of a jockey pump in each of the five ECCS loops is provided. Maintenance of the filled status of the system is ensured by continuous indication of pump operation and pump discharge pressure. In accordance with monthly surveillance procedures, the uppermost vent lines in the filled systems are opened and checked for flow to eliminate the possibility of the formation of air pockets. Pressure instrumentation provided on the jockey pump discharge line initiates an alarm in the main control room when pressure in the discharge line is less than the hydrostatic head required to maintain the line full of water up to the injection valves.

The emergency core cooling system pumps must have the capability to operate for an extended period of time during the long-term recirculating cooling phase following a loss-of-coolant accident. The acceptability of ECCS pumps capability is discussed in Section 3.11.

An electrical interlock is incorporated into the HPCS circuitry that prevents the injection valve from closing automatically upon receipt of the high reactor water level (L8) signals if a high drywell pressure signal still exists. The interlock was added as a result of the NRC staff review of GESSAR-238 which indicated that the interlock was needed to assure diversity of HPCS initiation signals and to prevent premature HPCS termination. However, flooding of the steam lines could result in damage to the safety/relief valves and primary system piping unnecessarily, since the interlock tends to keep the HPCS in operation past the point of reflooding the core and does not significantly add to the overall safety. We requested clarifications from the applicant regarding this item. We will report on this item in a supplement to this SER.

Safety relief valve operability will be demonstrated during plant startups by manually actuating each safety/relief valve (including the ADS valves) one at a time and observing the turbine bypass valve for change in position. The applicant stated that two designs are under consideration (a) direct valve position indications, via pressure switches in the SRV discharge line; (b) Acoustic sensors located on the discharge line. Either of these conceptual designs is acceptable to the staff but the applicant should document which has been selected prior to fuel load.

The staff asked the applicants to provide assurance that the safety relief valves have been qualified by environmental testing to support the assumption that 7 of the 8 ADS valves will operate. This is discussed in section 3.11 of this report.

6.3.3 Testing

The applicant states that operability of the emergency core cooling systems will be demonstrated by preoperational and periodic testing as required by Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants," and Criterion 37 of the General Design Criteria.

6.3.3.1 Preoperational Tests

Preoperational tests will assure proper functioning of controls, instrumentation, pumps, piping, and valves. Pressure differentials and flow rates will be measured for later use in determining acceptable performance in periodic tests. The applicant has committed to meet the guidelines of Regulatory Guide 1.68 mentioned above for preoperational and initial startup testing of the emergency core cooling system as noted in Section 14 of this report.

6.3.3.2 Periodic Component Tests

We will require the applicant to test the subsystems comprising the emergency core cooling system (except for the automatic depressurization system) every 92 days to show that specified flow rates are attained. Also, we will require every 18 months that a test be performed in which all subsystems are actuated through the emergency operating sequence. These tests comply with Criterion 37 of the General Design Criteria.

6.3.4 Performance Evaluation

We reviewed the loss-of-coolant accident analyses presented by the applicant in Section 6.3.3 of the Final Safety Analysis Report. Calculations were conducted in accordance with the methods described in General Electric Topical Report NEDO-20566, "General Electric Company Analytical Model for Loss-of-Coolant Analyses in Accordance with 10 CFR Part 50, Appendix K," dated August 1974 and "General Electric Refill Reflood Calculation" transmitted December 20, 1974. During 1977, the General Electric Company proposed several changes to its emergency core cooling system evaluation model. These changes have been approved by us and are described in our report "Safety Evaluation for General Electric Emergency Core Cooling System Evaluation Model Modifications." These methods constitute an evaluation model that conforms to the requirements of Appendix K to 10 CFR Part 50.

The five major acceptance criteria for the emergency core cooling system as specified in 10 CFR 50.46 are:

(1) The calculated maximum peak cladding temperature shall not exceed 2200°F.

(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) The calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core after any calculated successful initial operation of the ECCS.

At the time of this review, the FSAR LOCA analysis results were obtained for a lead plant representative of Perry. The licensee has committed to supply plant specific LOCA analysis in a later FSAR amendment. We will update the results as necessary after our receipt and review of the LOCA analysis.

The lead plant results for the first three items are:

	Maximum Values From Break Analyses	Allowable
Peak Cladding Temperature (PCT)	2066°F	2200°F
Maximum Cladding Oxidation	1.71%	17%
Maximum Total Hydrogen Generation	0.11%	1%

A coolable geometry is demonstrated by the compliance with the criteria for the PCT and the maximum cladding oxidation as discussed in NEDO-20566.

Long-term cooling is assured by the use of redundant systems which have adequate water sources available to remove the decay heat generated within the reactor core and transfer the heat to the ultimate heat sink. No single failure was identified that would prevent the ECCS from meeting this criterion. The systems are designed to reflood the reactor core to at least the 2/3 core level and maintain this level even under the most adverse circumstances. The major equipment for each system, other than the ADS, is located in separate water-tight rooms outside primary containment.

The applicant has indicated that some operator actions are assumed in the loss-of-coolant accident analyses 10 minutes after accident initiation. Section 6.3 of the Standard Review Plan states that no credit for operator actions should be taken prior to 20 minutes. We requested additional information from the applicant to confirm that no operator action is required until 20 minutes after the LOCA. We will report on this item in a supplement to this SER.

The applicant has not discussed the simultaneous closure of a recirculation flow control valve during a loss-of-coolant accident. We requested the applicant to address this concern. We will report our findings in a supplement to the SER.

The low pressure coolant injection flow may be diverted manually to drywell spray cooling or to suppression pool cooling. The Perry emergency procedures will contain adequate cautions to deter the operator from premature flow diversion. These procedures, which will be based on guidelines accepted by us (see SER Item I.C.1), caution the operator against diversion unless adequate core cooling is assured. LPCI diversion is identified in the guidelines as secondary to core cooling requirements except in those instances outside the design envelope involving multiple failures for which maintenance of containment integrity is required to minimize risk to the environment. We have reviewed the containment response analyses for the design basis event to determine the need for low pressure coolant injection diversion. These analyses indicate that there should be no need for wetwell spray actuation in the time frame during which the peak cladding temperature is reached. The operator's focus would, therefore, be on maintaining core cooling. Based on these analyses, the emergency procedures and guidelines discussed above, we find the applicants' position on low pressure coolant injection diversion to be acceptable. Review of all emergency procedures is being addressed in Items I.C.1 and I.C.8 of Chapter 22 of this report.

The core spray sparger for both the high and low pressure core spray systems each consists of two semicircular segments which form an essentially complete circular sparger. Water is sprayed radially onto the tops of the fuel assemblies by short elbow nozzles spaced around the sparger. Tests of this type of spray system were performed in a full-scale test in which air at atmospheric pressure simulated the post loss-of-coolant accident steam environment and indicated adequate cooling was delivered to each fuel assembly. However, recent tests conducted on a single nozzle indicate that the actual steam environment may adversely affect the distribution of flow from certain types of core spray nozzles. As discussed in NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," this problem is being studied by us under Task Action A-16 entitled, "Steam Effects on BWR Core Spray Distribution." Preliminary analyses and measurements have been made which support the existence of a significant safety margin between that amount of spray flow provided to each fuel assembly in the post loss-of-coolant accident steam environment and that used to calculate the spray cooling coefficients assumed in the loss-of-coolant accident analyses. Tests have recently been conducted by General Electric to confirm spray flow margins used in the emergency core cooling system loss-of-coolant accident analyses. We have reviewed the results of these tests and they have been found acceptable for BWR/6 plants. Our evaluation was forwarded to General Electric by letter dated January 30, 1981 from D. G. Eisenhut (NRC) to G. G. Sherwood (General Electric), "Acceptance for Referencing Topical Report NEDO-24712 Core Spray Design Methodology Confirmation Tests."

6.3.5 Conclusions

We reviewed piping and instrumentation drawings and the description of the emergency core cooling system presented in the Final Safety Analysis Report. We find the design of the system acceptable because it conforms to the pertinent Regulatory Guides, Standard Review Plan and General Design Criteria, Pending resolution of the issues noted above. In addition, based on the discussion above, we find the performance of the emergency core cooling system acceptable because it conforms with the requirements of 10 CFR 50.46 pending resolution of the issue noted above.

6.4 CONTROL ROOM HABITABILITY

Based upon our evaluation, the toxic gas and radiological consequences are within the acceptance criteria contained in SRP Section 6.4. Therefore, the staff finds that the design of the control room emergency ventilation system is acceptable for the purpose of preventing significant toxic gas and radiological exposure to operating personnel in the control room.

The control room design meets General Design Criterion 4, "Environmental and Missile Design Bases," with respect to "Structures, systems and components shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents..." This conclusion is based in part on the following:

No chlorine is stored on the site. Smoke detectors located in the control room air supply duct and in the emergency filter system discharge duct activate alarms to indicate the presence of smoke. The control room can be purged with outside air if necessary.

The two units at Perry share a common control building. The control room habitability system is sufficiently capable, diverse and redundant so that there is no impairment of the ability of the habitability systems to maintain a safe environment for control room personnnel during normal and accident conditions. Thus, the staff finds that the requirements of GDC-5, "Sharing of Structures, Systems and Components," have been met.

The applicant has protected the control room operators against radiation by the use of shielding and by the installation of a filtration system to remove airborne contaminants. After an accident, isolation occurs automatically in response to the accident signal (safety injection) or the high gaseous radioactivity signal for inlet air. This places the control room ventilation system in a recirculation mode with 30,000 cfm being circulated through redundant particulate and carbon filtration components.

In summary, the staff review was performed in accordance with Standard Review Plan Sections 2.2.1, 2.2.2, 2.2.3 and 6.4, and Regulatory Guides 1.78 and 1.95. The staff finds that the control room habitability system is adequate to provide safe, habitable conditions within the control room under both normal and accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

As a result, the staff concludes that the control room satisfies the requirements of (1) NUREG-0737, "Clarification of TMI Action Plan Requirements," issued in November 1980, and (2) GDC-19, "Control Room," and is, therefore, acceptable for a full power license.

6.5.3 FISSION PRODUCT CONTROL SYSTEMS

The applicant has designed three systems, an annulus exhaust gas treatment system (AEGTS), a main steam line isolation valve leakage collection system (LCS), and a fuel handling area exhaust system (FHAES) to control the leakage of airborne radioactivity following accidents. The AEGTS is capable of filtering and exhausting air from the secondary containment, which encloses the areas surrounding containment, and the pipes and equipment which may serve as pathways for the release of radioactivity. The LCS is capable of exhausting gas from the main steam lines into volumes served by the AEGTS, thus preventing those lines from leaking to the turbine building. The FHAES is designed to exhaust air from the fuel handling area following a fuel handling accident.

The secondary containment volume will be kept at subatmospheric pressure by its normal ventilation system. Upon activation, the AEGTS will replace the normal ventilation system and maintain a subatmospheric pressure. The AEGTS has two redundant trains, each having a capacity of 2000 cfm, and charcoal and particulate filters sufficient to remove more than 99%

of any gaseous iodine species and particulate material in the air it processes.

The AEGTS is provided to reduce the quantity of fission products released to the environment following postulated accidents and provides suitable redundancy in components and features such that its safety function can be accomplished assuming a single failure. Thus, the system conforms to General Design Criterion 41, "Containment Atmosphere Cleanup."

The AEGTS is designed to permit periodic inspection and testing and, therefore, conforms to General Design Criterion 42, "Inspection of Containment Atmosphere Cleanup Systems" and General Design Criterion 43, "Testing of Containment Atmosphere Cleanup Systems." We conclude that the AEGTS design is acceptable.

The LCS consists of two trains which can draw suction from either the volume between the main steam line inboard and outboard isolation valves or from the volume beyond the outboard isolation valve. Any leakage through the main steam line isolation valves is drawn by this system into the secondary containment which is served by the AEGTS. The operation of the LCS following an accident would prevent isolation valve leakage from passing into the turbine building through the main steam lines. The collected leakage would be delayed, diluted and filtered to remove most particulates and gaseous iodine species, prior to release by the AEGTS.

The LCS is actuated manually, but is protected by an interlock from opening valves leading from the main steam line if the steam line pressure is too high. Upon actuation, the system will attempt to establish suction to the main steam line between the two isolation valves. If an excessive amount of steam exists in this volume, due to failure of the inboard isolation valve to close, the LCS will automatically exhaust the steam line beyond the outboard isolation valve.

The system has the capacity to operate as long as at least one of the two isolation valves and at least one of the slow-closing downstream valves in each steam line closes. The applicant proposes to manually actuate the LCS approximately 20 minutes after the occurrence of a loss-of-coolant accident. Our evaluation of this potential accident indicates that of the order of a few hours would be required for the transit of fission gases through the isolated portion of the main steam lines, assuming single failure of any isolated valve. Therefore, ample time for manual actuation exists and automatic actuation is not required.

The applicant has provided an LCS that is designed in accordance with the regulatory positions set forth in Regulatory Guides 1.96, and which meets General Design Criterion 54, "Piping Systems Penetrating Containment". The staff finds the design of this system acceptable.

The general fuel handling areas, fuel pool areas and the fuel pool cooling equipment rooms are ventilated by the fuel handling area supply system (FHASS) and the fuel handling area exhaust system (FHAES).

Radiation monitoring is also provided to alarm in the control room if the radioactivity level in the exhaust air exceeds a preselected set point. The radiation monitor is located in the ventilation exhaust duct at the main exhaust header, and draws representative samples of air from the area.

The FHAES continuously draws air from the CRD pump areas, above the fuel pools, and from the fuel pool cooling and cleanup equipment rooms located in the intermediate building. Two of the three 50 percent charcoal exhaust units are operating normally to draw air through the exhaust ductwork and discharge it to the atmosphere through the unit vent.

In the event that the radiation monitors located upstream of the charcoal exhaust units sense high radiation, the high radiation signal alarms in the control room and automatically trips the supply fan. The exhaust system remains operational to continue exhausting contaminated air from the fuel handling area through charcoal filters, thus precluding any uncontrolled release of radioactivity to the outside. We find these design provisions acceptable.

EFFLUENT TREATMENT SYSTEMS BRANCH

6.5.1 ESF ATMOSPHERE CLEANUP SYSTEMS

The review performed under Standard Review Plan 6.5.1 pertains to the filtration and cleanup systems provided by the applicant for the purposes of (1) controlling the releases of radioactive material in plant gaseous effluents (radioiodine and particulate material) and (2) controlling the concentrations of radioactive material in the recirculated control room atmosphere within habitable limits following a design basis accident (DBA). In the Perry Nuclear Power Plant, Unit No. 1, design, there are three filtration systems designed for these purposes: (1) the Control Room Emergency Recirculation System (CRERS); (2) the Fuel Handling Area Charcoal Exhaust System (FHACES); and (3) the Annulus Exhaust Gas Treatment System (AEGTS). Our review was concerned with the design of system components, design features which influence system availability and reliability, and the design efficiency of media installed in the atmosphere cleanup systems for the removal of radioactive materials from the process or effluent stream.

I. DESIGN OF ESF ATMOSPHERE CLEANUP SYSTEMS

The CRERS, FHACES, and AEGTS are safety related systems. The system designs conform to the requirements of GDC 1, 2, 3, 4, 19, 60, and 61 of Appendix A to 10 CFR Part 50. The guidance in Regulatory Guide 1.52 has been considered in the design of these systems. One deviation from full conformance with

R.G. 1.52 has been noted; however, the deviation was considered acceptable design practice at the design stage and, therefore, continues to be acceptable to the staff at this time. Position C.5.k of R.G. 1.52 and ANSI N509-1976 recommend that means be provided for preventing possible iodine desorption and adsorbent autoignition that may result from radiation-induced heat in the absorbent and any accompanying temperature rise. Acceptable designs include low-flow air bleed systems, cooling coils, water sprays for the adsorber section, or other cooling mechanisms. Rather than a cooling system to prevent autoignition, the applicant has provided a water deluge spray system for flooding the adsorbers to extinguish fires in the adsorbers; the system is manually actuated from the control room on the basis of high temperature alarms from sensors within the charcoal beds. Design of the CRERS, FHACES, and AEGTS is to seismic Category 1 and to pertinent sections of ANSI N509 and ERDA 76-21, as outlined in Regulatory Guide 1.52. Redundant trains are provided for each system. Charcoal adsorbers are of the integral type, with top loading and bottom unloading of bulk charcoal.

II. TESTING OF ESF ATMOSPHERE CLEANUP SYSTEM COMPONENTS

Testing of ESF atmosphere cleanup systems and components is in accordance with the provisions of Regulatory Guide 1.52.

- Air flow distribution is in accordance with Position C.5.b of Regulatory Guide 1.52.
- In-place testing of HEPA filter section is in accordance with ANSI N510; however, testing criteria do not make mention of portions of Position C.5.b of Regulatory Guide 1.52, e.g., specific intervals or following painting, fire, or chemical release (may be open item or, alternatively, may be in Tech Specs).
- Leak tests of charcoal adsorber sections to use halogenated hydrocarbon refrigerant gas.
- Test canisters will be used for determination of radio-halogen retention.

460.4 OPEN ITEMS: Does not mention testing requirements for specific intervals between tests and does not specify testing following painting, fire, or chemical release.

III. INSTRUMENTATION REQUIREMENTS - ESF CLEANUP SYSTEMS

A. CRERS

Instrumentation provided for the CRERS includes:

- Flow rate, unit outlet. Provides indication and high/low alarms in the main control room. No provision for recording flow as recommended in ANSI 509 or Regulatory Guide 1.52.

- Temperature, charcoal bed. Provides indication and high/low alarms in the main control room. No requirements specified in Regulatory Guide 1.52.
- Pressure Drop (Δp). Provides indication, status of operation, and alarms in main control room. Components covered include roughing filter, upstream HEPA filter, charcoal bed, and downstream HEPA filter. No provision for recording of any system pressure drops. No provisions for measurement of total pressure drop across complete system. Regulatory Guide 1.52, Section C.2.g, recommends recording of "pertinent" pressure drops at the control room.

460.5 OPEN ITEMS: No provision for control room recording of system
460.6 air flow rates or pressure drops. No provision for measurement, indication, or recording of total system pressure drop ($\sum \Delta p$).
460.7 No provision for status indication in control room of deluge valve positions, valve/damper operator position, or fan status.

B. AEGTS

Instrumentation provided for the AEGTS includes:

- Low Air Flow. Alarm in control room.
- Pressure Drop (Δp). Indicator(s) in control room. (Note: Not specific in list, page 7.3-35)
- Temperature. Indicators in control room (Charcoal bed).

- Radiation. Indicator in control room.

(Note: P&ID shows Δp sensors on prefilter, upstream HEPA charcoal bed, and downstream HEPA; these are not specified in list on page 7.3-35.)

460.8, 460.9 OPEN ITEMS: Total system$\Sigma\Delta$p not provided. No provision for indication and recording of system flow rate in control room.

460.10 Section 6.5.1.6, FSAR, references Section 6.5.3 for instrumentation and actuation requirements; correct reference is Section 7.3.1.

C. FHACES/FHAES

Instrumentation provided for the FHACES includes (Ref. Section 7.6.1.9b):

- Fan Status. Status light in control room.
- Low Air Flow. Alarm in control room.
- Smoke. Alarm in control room
- High/Low Alarm, Heating Coil Air. Alarm in control room.
- High Radiation, in Exhaust Duct. Alarm in control room.
- Charcoal Bed Temperature. Continuous indication on control room panel high/high alarms.
- Moisture, Exhaust Air. High alarm in control room.

OPEN ITEMS: Items not consistent with ANSI N509 and Regulatory Guide 1.52:

460.11 - Unit outlet flow not shown to be indicated or recorded in control room.

460.12 - Component or system pressure drops ($\Delta p, \Sigma\Delta p$) not indicated or recorded in control room.

460.13 - No status indication in control room of deluge valve position, valve/damper operator position.

460.14 - Section 6.5.1.5, FSAR, references Section 9.4.1 for instrumentation and actuation requirements; correct reference is Section 7.6.1.9.

IV. FINDINGS

Subject to resolution of the above open issues, our findings are as follows:

The ESF atmosphere cleanup systems include the equipment and instrumentation to control the release of radioactive materials in gaseous effluents following a postulated design basis accident. The scope of our review included an evaluation of these systems with respect to the guidelines of Regulatory Guide 1.52. We have reviewed the applicant's system descriptions and design criteria for the ESF atmosphere cleanup systems. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the ESF cleanup systems to applicable regulations, guides and industry standards. Based on our evaluation, we find the proposd ESF atmosphere cleanup systems are acceptable, and the filter efficiencies given in TAble 2 of Regulatory Guide 1.52 are appropriate for use in accident analyses.

6.7 Main Steam Isolation Valve Leakage Control System

The main steam isolation valve leakage control system (MSIVLCS) is designed to control and minimize the release of fission products which could leak through the closed MSIVs after a loss-of-coolant accident. The system consists of two separate and redundant subsystems. One subsystem functions to maintain the steam lines between the MSIVs at a slight vacuum following system actuation. The other subsystem functions to maintain the steam lines between the outboard MSIVs and the main steam line shut-off valves at a slight vacuum following system actuation. Each subsystem receives power from a separate division of the emergency power supply. Both subsystems are actuated manually and simultaneously and both exhaust to the annulus between the containment and shield buildings served by the annulus exhaust gas treatment system for processing prior to release to the atmosphere.

The operation of the system is limited by a series of pressure sensors and timers which serve as interlocks designed to preclude system actuation prior to the pressure in the main steam lines decreasing to the pressure for which the leakage control system is designed to operate. The interlocks also preclude continued operation of any portion of the leakage control system which fails to achieve a subatmospheric condition in its respective steam line after a preset time. In addition, an interlock is provided to prevent operation of an individual inboard main steam isolation valve leakage control system unless the corresponding main steam isolation valve inside the containment is fully closed. The main steam isolation valve leakage control system will be manually initiated no sooner than twenty minutes following a postulated design basis loss-of-coolant accident in accordance with Regulatory Guide 1.96, "Design of Main Steam Isolation Leakage Control Systems for the Boiling Water Reactor Nuclear Power Plants." The required actuation time period will be consistent with loading requirements on the emergency electrical buses, with reasonable times for operator information, decision, and action, and will be consistent with the time required for main steam line pressure decay following a postulated loss-of-coolant accident.

The applicant has proposed a technical specification which provides an allowable MSIV leak rate of 25 scfh per valve as this value was assumed in the accident analysis (refer to Section 15.3 of this SER for our review of the accident analysis). Because our review of the offsite dose contribution based on 25 scfh MSIV leakage is acceptable, we find the proposed MSIV leak rate of 25 scfh acceptable in lieu of the 11.5 scfh standard technical specification MSIV allowable leak rate. MSIV leak rate verification test frequency and MSIVLCS test frequency will be in accordance with the standard technical specification.

Based on the above, we conclude that the requirements of General Design Criterion 54, "Piping Systems Penetrating Containment" and the guidelines of Regulatory Guide 1.96 with respect to functional design are satisfied.

The system is located in a seismic-Category-I, flood- and tornado-protected structure (refer to Sections 3.4.2 and 3.5.2 of this SER). The MSIVLCS itself is seismic Category I. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.29, "Seismic Design Classification," and 1.96 are satisfied.

Since the system would be called on to function only in the event of a Loss-of-Coolant Accident (LOCA), it is capable of performing its safety function under the expected LOCA environmental conditions appropriate to the system equipment location. (Refer to Section 3.11 of this SER). Further, the components of each subsystem are protected by separation and barriers against internally generated missiles, externally generated missiles, and dynamic effects associated with pipe breaks, such that their function is not impaired under postulated LOCA conditions. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.96 are satisfied.

Based on our review, we conclude that the MSIVLCS is in conformance with the requirements of General Design Criteria 2 and 4 and the guidelines of Regulatory Guides 1.29 and 1.96 with respect to protection against natural phenomena, missiles, pipe break effects and seismic classification, and with requirements of General Design Criterion 54 and the guidelines of Regulatory Guide 1.96 relating to functional design and is, therefore, acceptable.

INSTRUMENTATION AND CONTROLS

SECTION 7

ELECTRIC POWER

SECTION 8

AUXILIARY SYSTEMS

SECTION 9

AUXILIARY SYSTEMS

OPEN AREAS

Section-9.1.2 -- Spent Fuel Storage-- The applicant is required to describe the provisions for monitoring and alarming excessive fuel pool liner leakage.

Section 9.1.3-Spent Fuel Pool Cooling and Cleanup System -- The applicant is required to provide a technical specification that prevents reactor startup when the RHR system is providing spent fuel cooling.

Section 9.1.4 Light Load Handling System (Relating to Refueling) -- The applicant is required to verify that the auxiliary hook of the fuel handling area crane and the auxiliary hoist of the fuel handling platform are designed to seismic Category I requirements.

Section 9.2.1 Station Service Water System -- The applicant is required to describe the plant provision to monitor possibe flow blockage in the emergency service water system resulting from sources such as Asiatic clams.

Section 9.3.1 Compressed Air System -- The applicant is required tp provide a commitment to test the instrument air quality at least yearly. The test should verify proper air quality at the dryer-filter discharge and atn the end of each air line branch header. Additionally, the applicant is required to provide the basis for the exception to the air quality standard concerning maximum particle size.

Section 9.3.3 Equipment and Flood Drainage System --

1. The applicant is required to verify that each ECCS pump room has redundant, safety grade level switches to alarm in the control room in case of leakage from equipment in the room or demonstrate that a single non-safety grade switch is qualified to perform this function.

2. The applicant is required to describe the methods that will be used to assure that the manually operated valve in the drain line for each ECCS pump room will be closed during normal plant operation. Additionally, the applicant is required to describe the methods for preventing ~~XXX~~ back flooding following an SSE that would cause failure of non-seismic piping system.

Section 9.4.1 Control Room Area Ventilation System --

1. We note that parts of Tables 3.2-1, "Classification of Systems, Components and Structures", are incomplete. In Section XXXV, Heating, Cooling and Ventilation Systems, the applicant is required to expand Items 11, 12, 17, 18, 20, 24, 25, 26 and 27 to show a) fans and motors, b) cooling coils, c) filters, d) humidifiers, e) charcoal filter housing, f) ductwork and dampers, g) valves with safety isolation function, h) electrical and unit heaters, i) electrical modules with safety function, j) cable with safety function, as applicable to a given system. Additionally, the applicant is required to expand Items 31 and 32 to show all equipme t such as motors, pumps, heat exchangers, piping, valves and electrical modules and cable with a safety function.

2. Table 6.4-4, Control Room Emergency Filter System, Single Failure Analysis, covers the emergency filtration subsystem only. The applicant is required to provide a single failure analysis for the Control Room HVAC System.

3. In Section 6.4.2.3, Leak Tightness, the FSAR states that the inlet ducts contain two isolation dampers in series. These are shown in Figure 6.4-1, the P&ID for the system. The FSAR further states that the exhaust ~~XXXX~~ isolation dampers are similar in arrangement to the inlet dampers. However, the P&ID shows only one damper per exhaust duct. The applicant is required to confirm that there are two dampers in series in each exhaust duct and correct Figure 6.4-1 accordingly.

4. In Section 2.2.3.1.2.1, The FSAR states that chlorine and ethylene oxide detectors will automatically initiate isolation of the control room upon detection of their respective gases. These detectors are also not shown in Figure 6.4-1. Radiation detectors are also not shown on the inlet ducts. The applicant is required to revise Figure 6.4-1 to show redundant chlorine, ethylene oxide and radiation detectors in each of the inlet ducts that will automatically initiate isolation of the conrol room upon detection.

5. In Section 6.4, the FSAR states that the control room HVAC system controls, alarms, readout instruments, etc. are located in the control room. Since there are two control rooms, and presumably two sets of operators --one for each unit, the applicant is required to describe where the panel(s) for this system is (are) located and discuss the method of operation of the shared system under all modes of station operation.

6. Figure 1.2-6 of the FSAR shows the six emergency diesel exhaust silencers on the roof of the diesel generator building at an elevation of approximately 650 ft. and approximately 75 ft. west of the control building wall. Figure 1.2-9 shows two air intakes on the west wall of the control,building at the approximate elevation of 680 ft. It also shows air intakes on the north and south walls of the control building at the same elevation. Describe the systems serviced by these air intakes. In the event of a west wind simultaneous with the diesel operation, it seems possible that poisonous diesel exhaust fumes could be drawn into the control building. The applicant is required to demonstrate either that this cannot occur, or that it would not pose a threat to the safe conduct of operation of the station.

Section 9.4.3 Auxiliary and Radwaste Area Ventilation System --

1. The applicant is required to describe the means provided to assure that the temperature in the room housing the spent fuel pool cooling pumps, the room housing the hydrogen recombiner equipment, the rooms housing the annulus exhaust gas treatment system trains, and the rooms housing the spent fuel pool area ventilation system exhaust trains can be maintained at acceptable levels for equipment operation under accident and emergency conditions when the normal intermediate building ventilation system is not operating.

2. Section 9.4.3 and Section 3.5.2 -Structures, Systems and Components to be Protected from Externally Generated Missiles- The applicant is required to describe the means provided to prevent tornado,missiles from entering the intermediate building through the intermediate building ventilation system air inlet.

3.
Section 9.4.5 Engineered Safety Feature Ventilation System-- The applicant is required to provide the ambient conditions required for operation of the hydrogen recombiner equipment located outside of containment. Additionally, indicate the safety-related system that will maintain these conditions following a LOCA and loss of offsite power.

9.0 Auxiliary Systems

We have reviewed the design of the auxiliary systems necessary for safe reactor operation, shutdown, fuel storage or whose failure might affect plant safety including their safety related objectives, and the manner in which these objectives are achieved.

The auxiliary systems necessary for safe reactor operation or shutdown include the emergency service water system, the emergency closed cooling system, the ultimate heat sink, the heating ventilation and air conditioning systems for the control room and areas housing safety-related equipment, essential portions of the compressed air system, and the standby liquid control system.

The auxiliary systems necessary to assure the safety of the fuel storage facility include new fuel storage, spent fuel storage, the spent fuel pool cooling and cleanup system, fuel handling systems and the fuel handling area heating, ventilation and air conditioning system.

We have also reviewed other auxiliary systems to verify that their failure will not prevent safe shutdown of the plant or result in unacceptable release of radioactivity to the environment. These systems include: the nuclear closed cooling water system, the demineralized water makeup system, potable and sanitary water system, the condensate storage facilities, the turbine building closed cooling water system, the chilled water systems, non-essential portions of the compressed air system, the equipment and floor drainage system, and heating, ventilation, and air conditioning systems for non-essential portions of the auxiliary building, intermediate building, the radwaste building, and the turbine building.

9.1 Fuel Storage Facility

9.1.1 New Fuel Storage

The new fuel storage facility consists of two separate new fuel storage vaults located in the shared intermediate building. The facility provides dry storage for a maximum of 360 fuel assemblies (180 per vault, 24% of a core load) and includes the new fuel assembly storage racks and the concrete storage vaults that contain the storage racks. Sharing of the storage area between Units 1 and 2 does not impair essential plant safety systems since such sharing or a single failure occurring in the storage area does not prevent safe shutdown of either or both units nor does it significantly increase the potential for radioactive releases. Therefore the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are satisfied.

The intermediate building which houses the facility, and the storage rack and vaults are designed to seismic Category I criteria. This building is also designed against flooding and tornado missiles (refer to Sections 3.4.1 and 3.5.2 of this SER). Thus, the requirements of General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," are satisfied.

The vaults housing the new fuel storage racks are not located in the vicinity of any moderate- or high-energy lines or rotating machinery. This separation from such potential missile sources protects the new fuel from internally generated missiles and the effects of pipe breaks (refer to Sections 3.5.1.1, 3.5.1.2 and 3.6.1 of this SER).

The facility is designed to store unirradiated, low emission fuel assemblies. Accidental damage to the fuel would release relatively minor amounts of radioactivity that would be accommodated by the spent fuel pool area ventilation system. Thus, the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," are satisfied.

The new fuel storage racks are designed to store the fuel assemblies in an array which is sufficient to maintain a K_{eff} of 0.95 or less in the normal dry condition or abnormal completely water flooded condition. The racks are not designed to maintain a K_{eff} of 0.98 or less under optimum moderation (foam, small droplets, spray or fogging). The condition of optimum moderation is precluded since the new fuel storage vault is provided with solid watertight cover. The applicant will utilize adminsitrative controls to preclude entry of sources of optimum moderation into the new fuel storage area during movement of fuel, thereby significantly reducing the probability of such a condition. In addition, the floor of the vault is sloped to a drain to remove any water introduced into the vault. We find this approach acceptable. The racks themselves are designed to preclude the inadvertent placement of a fuel assembly in other than the prescribed spacing. Thus, the requirements of General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling are satisfied.

Based on our review, we conclude that the new fuel storage facility is in conformance with the requirements of General Design Criteria 2, 5, 61 and 62 as they relate to new fuel protection against natural phenomena, missiles, shared functions, radiation protection and prevention of criticality, and the guidelines of Regulatory Guide 1.29 relating to seismic classification and is, therefore, acceptable.

9.1.2 Spent Fuel Storage

A shared spent fuel storage facility is provided for the plant. Low density storage racks located in the upper containment pool of each unit have a capacity of 190 fuel assemblies (25% of a full core) per unit. The shared intermediate building contains two shared pools, the fuel preparation and storage pool and the storage pool. These pools contain high density storage racks with a capacity of 4020 fuel assemblies (537% of a full core) for both units.

The structures housing the facility (the intermediate building and containments) are designed to seismic Category I criteria as are the storage racks, pool liners, gates and storage pools. These buildings are also designed against flooding and tornado missiles (refer to Sections 3.4.1 and 3.5.2 of this SER). We conclude that the requirements of General Design Criterion 2, "Design Bases for Protection Against natural Phenomena," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," 1.29, "Seismic Design Classification," and 1.117, "Tornado Design Classification," are satisfied for the spent fuel storage facility.

The fuel pools are not located in the vicinity of any high-energy lines or rotating machinery. Therefore, physical protection by means of separation is utilized to protect the spent fuel from internally-generated missiles and the effects of pipe breaks (refer to Sections 3.5.1.1 and 3.6.1 of this SER). Thus the requirements of General Design Criterion 4, "Environmental and Missile Design Bases" and the guidelines of Regulatory Guides 1.13 are satisfied. For a discussion of compliance with the guidelines of Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles" refer to Section 3.5.1.3 of this SER.

The shared portion of the facility has sufficient redundancy of services and is of seismic Category I, Quality Group C design, so that an accident in one unit with loss of offsite power will not impair its ability to safely store the spent fuel. This satisfies the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components."

The low-density and high-density storage areas of the facility are designed to store the fuel assemblies in an array which limits K_{eff} to 0.95 or less. The low-density storage racks are aluminum with a fuel assembly minimum center-to-center storage spacing of 7 inches. The high-density storage racks are aluminum with a neutron poison material between storage spaces and provide a fuel assembly minimum center-to-center storage spacing of 6-5/8 inches. The racks are designed to preclude the inadvertent placement of a fuel assembly in other than the prescribed spacing. The racks can withstand the impact of a dropped fuel assembly without unacceptable damage to the fuel and can withstand the maximum uplift forces exerted by the fuel handling machine. Thus, the requirements of General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control," and 62, "Prevention of Criticality in Fuel Storage and Handling," and the guidelines of Regulatory Guide 1.13 concerning fuel storage facility design are satisfied.

Based on our review, we conclude that the spent fuel storage facility is in conformance with the requirements of General Design Criteria 2, 4, 5, 61, and 62 as they relate to protection of spent fuel against natural phenomena, missiles, environmental effects, the facility's shared functions, radiation protection, and prevention of criticality and the guidelines of Regulatory

Guides 1.13, 1.29, and 1.117 relating to the facility's design, seismic classification, and protection against tornado missiles.

[However, provisions for monitoring and alarming excessive pool liner leakage have not been described, and we cannot determine if the requirements of General Design Criterion 63, "Monitoring Fuel and Waste Storage" have been met. Until satisfactory compliance with this criterion is determined, the spent fuel storage facility is unacceptable.

The applicant is required to describe the provisions for monitoring and alarming excessive fuel pool liner leakage to assure complaince with the requirements of General Design Criterion ~~6.3~~ 63.] We will report on the resolution of this matter in a supplement to this SER.

SAFETY EVALUATION REPORT
CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2
CHEMICAL ENGINEERING BRANCH

..2 Spent Fuel Storage (cont'd)

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a subcritical array during all credible storage conditions. We have reviewed the compatibility and chemical stability of the materials (except the fuel assemblies) wetted by the pool water in accordance with Standard Review Plan 9.1.2.

There are two types of spent fuel storage facilities in the Perry Plant: a small pool in each containment building with capacity to store 25% of the spent fuel assemblies of a full core, and a larger pool in the intermediate building with storage capacity for 4020 spent fuel assemblies. For both types of storage, the racks are fabricated of aluminum and the pools are lined with 304 stainless steel. Demineralized oxygen-saturated water is used for cooling the pool.

In the small storage pool, the aluminum racks are bolted to the stainless steel liners, and subcriticality is achieved by a loose-packed geometric array.

In the large spent fuel storage pool, the fuel assemblies are more closely packed, with criticality controlled by sheets of neutron absorber (Boral) between adjacent fuel assemblies. Boral contains natural boron carbide in an aluminum matrix clad with 1100 Series aluminum. The aluminum fuel racks are free-standing on legs provided with leveling screws. The aluminum legs are insulated from the stainless steel liner by plastic pads.

Evaluation

In the environment of oxygen-saturated high purity water, the anticipated corrosion rate of the stainless steel and aluminum alloys located in the pool is negligible. The corrosion rate of aluminum in water of pH 7 at 125° C is $1.5X10^{-4}$ mils/day (1.1 mils per 20 years)[1]. Corrosion rates of this order are not of practical concern and even lower rates are expected under actual service conditions.

Experience has shown that galvanic couples between stainless steel and aluminum do not give rise to significant localized corrosion in BWR spent fuel pool environments, since the metals are protected by highly passivating oxide films and therefore at similar potentials in pure water.[2]
The high density fuel racks are further protected by an anodized surface and by plastic pads insulating the aluminum legs from the stainless steel liner. The potential for galvanic corrosion is therefore negligible.

Conclusion

On the basis that the alloys used in constructing the spent fuel storage pools have a high resistance to general corrosion and localized corrosion, have a low differential galvanic potential between them, and have been shown by test data and service experience in operating reactors to be compatible with the BWR spent fuel pool environment, we conclude that the corrosion that will occur in the spent fuel pool should be of little significance during the 40-year life of the plant.

We therefore find that the selection of appropriate materials of construction by the lecensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61, having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, preventing criticality by maintaining structural integrity of components and of the boron poison. The selection of materials of construction is therefore acceptable.

References

1. J. E. Draley and W. E. Ruther, Report No. ANL-5001, February 1953.

2. J. R. Weeks, Corrosion of Materials in Spent Fuel Storage Pools, BNL-NUREG 23021, July 1977.

requirements of ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities." These measures demonstrate their suitability to withstand a postulated design basis accident (DBA) environment.

The control of combustible gases that can potentially be generated from the organic materials and from qualified and unqualified paints is reviewed under Section 6.2.5. The consequences of solid debris that can potentially be formed from unqualified paints are reviewed under Section 6.2.2.

*9.1.3 Spent Fuel Pool Cleanup System

I. INTRODUCTION

The spent fuel pool cleanup system is designed to maintain optical clarity and to remove corrosion products, fission products, and impurities from the spent fuel pool water.

* This evaluation is based upon applicant information through FSAR amendment 3 and by telecon and telex and is subject to confirmatory review of committed FSAR amendments.

Water purity and clarity in the storage pool, reactor well, and dryer-separator pit are maintained by filtering and demineralizing the pool water through a filter demineralizer. In addition to fuel pool water demineralization, the system will be used on occasion to demineralize suppression pool water.

The pool cleanup system consists of a bypass flow loop around the pool cooling system pumps with two 100% parallel filter demineralizers, and the required piping, valves, and instrumentation.

Continuous influent and effluent conductivities for the fuel pool demineralizers are monitored and recorded. A high conductivity effluent alarm setpoint of 1.5 umho/cm is chosen to reflect marginal performance of the demineralizers since they will eventually fill with air saturated water at an equilibrium level of about 1.1 umho/cm. Differential pressure drop is continusously monitored across the filter demineralizers and the units are removed from service for recoating with a combined filtration/ ionexchange media if the conductivity limits or differential pressure set points are exceeded.

Weekly fuel pool analyses will be performed to assure that the NSSS water quality specifications for the fuel pool are maintained. The water quality parameters are as follows:

Conductivity	< 3 umho/cm at 25°C
Chloride	< 0.5 ppm
pH	5.3 - 7.5 at 25°C
Total Insolubles	<1 ppm
Heavy metals	<0.1 ppm

Weekly gross gamma analyses are performed following fuel load or when spent fuel is stored in the pool. Special tests on ioding or other significant radionuclides removed by the fuel pool filter demineralizer will be peformed when gross gamma activity levels in the fuel pool exceed 2000 CPM/ML during normal power operation.

II. EVALUATION AND FINDINGS

The system description and piping and instrument diagram were reviewed in accordance with SRP Section 9.1.3. We determined that the fuel pool purification system (1) provides the capability and capacity of removing radioactive materials, corrosion products and impurities from the pool water, and thus meets the requirements of GDC 61 in Appendix A to 10 CFR Part 50, as it relates to appropriate filtering systems for fuel storage;

(2) is capable of reducing occupational exposure to radiation by removing radioactive products from the pool water, and thus meets the requirements of Section 20.1(c) of 10 CFR Part 50, as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confines radioactive materials in the fuel pool water within the demineralizer and filters, and thus meets Regulatory Position C.2.f(2) of Regulatory Guide 8.8, as it relates to reducing the spread of contaminants from the source; and (4) removes suspended impurities from the pool water by filters, and thus meets Regulatory Position C.2.f(3) of Regulatory Guide 8.8, as it relates to removing crud through physical action.

On the basis of the above evaluation, we find that the spent fuel pool cleanup system meets GDC 61, Section 20.1(c) of 10 CFR Part 20 and the appropriate section of Regulatory Guide 8.8 and, therefore, is acceptable.

9-11

*9.3.2 Process Sampling System

I. INTRODUCTION

The process sampling system is designed to provide

* This evaluation is based upon applicant information through FSAR amendment 3 and by telecon and telex and is subject to confirmatory review of committed FSAR amendments.

9.1.3 Spent Fuel Pool Cooling and Cleanup System (cont'd)

The spent fuel pool cooling and cleanup system is designed to maintain water quality and clarity and remove decay heat generated by spent fuel assemblies in the pool. The system includes all components and piping from inlet to exit from the storage pools, piping used for fuel pool makeup, and the cleanup filter/demineralizers to the point of discharge to the radwaste system. The design consists of two fuel pool cooling pump/heat exchanger trains and two sets of filter/demineralizers. Each fuel pool cooling pump can be powered from a redundant division of the Class 1E power system.

The system is housed in the intermediate building and containment (seismic Category I and tornado protected structures). The system itself, with the exception of the cleanup portion, is designed to Quality Group C and seismic Category I requirements. In case of a seismic event, a seismic Category I bypass line and redundant seismic Category I isolation valves have been provided at the cleanup system connections to the fuel pool cooling lines to isolate the non-seismic Category I portion of the system to assure that failure in that portion of the system has no adverse effect on safety-related equipment. This design satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of

Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," and 1.29, "Seismic Design Classification."

The various components of the system are located in shielded cubicles or are separated from other moderate and high-energy piping systems. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.13 are satisfied.

The system serves the shared fuel storage facility with sufficient redundancy of equipment and conservative design that the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are met.

The system is accessible for routine visual inspection of the system components. One fuel pool cooling pump is in operation at all times. The spare pump will be operated periodically in accordance with plant Technical Specifications. Thus, the requirements of General Design Criterion 45, "Inspection of Cooling Water Systems," and 46, "Testing of Cooling Water Systems," are satisfied.

The spent fuel pool cooling system will maintain the fuel pool water temperature at 127°F with a heat load based on decay heat generation from 4020 fuel assemblies (maximum storage) and both cooling trains in operation. This is the normal discharge from nine years of operation of both units. If one pump and heat exchanger were lost under these conditions, the temperature would rise to 160°F maximum for a short period. The applicant has committed to running the RHR system of a shutdown reactor to maintain temperatures below 150°F until the normal system could maintain the temperature below 150°F.

In the case of an abnormal heat load when the full core of one unit must be unloaded, the RHR system of the affected unit would be used to maintain the temperature of the fuel pool below 150°F. Under these conditions, the RHR system could maintain a fuel pool temperature of 106°F. Again, the RHR system would be used until the normal system could maintain the temperature below 150°F. The reactor of the unit whose RHR system is providing pool cooling, will not be started up. [It is our position that a technical specification be provided to prevent reactor startup when the RHR system is providing fuel pool cooling.]

Heat loads for the above storage modes are based on Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling."

No connections are provided to the spent fuel pool that may cause the pool water to be drained below a safe shielding level. All lines that connect to the pool and extend below the safe level of the pool water are equipped with syphon breakers, check valves or other means to prevent inadvertent pool drainage. The nonsafety-related nuclear closed cooling water system provides cooling water to the fuel pool heat exchanger under normal conditions. Backup cooling is available in emergency conditions from the emergency (seismic Category I) closed cooling water system which transfers spent fuel pool heat loads to the ultimate heat sink (refer to Sections 9.2.1, 9.2.2, and 9.2.5 of this SER). In addition, the residual heat removal system can be utilized to supplement the fuel pool cooling system by providing additional cooling during shutdown as described above. Thus, the requirements of General Design Criterion 44, "Cooling Water," are met.

Normal makeup to the pool is provided by the nonsafety-related condensate and refueling water storage and transfer system to replace losses due to leakage through the liner and evaporation. Emergency makeup is supplied by the redundant loops of the seismic Category I emergency service water system. Thus, the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guide 1.13 concerning fuel pool design are satisfied.

The system incorporates control room alarmed pool water level, water temperature, and building radiation level monitoring systems, thus satisfying the requirements of General Design Criterion 63, "Monitoring Fuel and Waste Storage."

Based on our review, we conclude that the spent fuel pool cooling and cleanup system is in conformance with the requirements of General Design Criteria 2, 4, 5, 44 45, 46, 61 and 63 as they relate to protection against natural phenomena, missiles, and environmental effects, shared functions, cooling water capability, inservice inspection, functional testing, radiation protection and monitoring provisions, and the guidelines of Regulatory Guides 1.13, and 1.29, and Branch Technical Position ASB 9-2 relating to the system's functional design, seismic classification, and design decay heat load, and is, therefore, acceptable, [pending the applicant's commitment to provide a technical specification preventing reactor startup when the RHR system is providing fuel pool cooling]. We will report on the resolution of this matter in a supplement to this SER.

9.1.4 Light Load Handling System (Related to Refueling)

The fuel handling system provides the means of transporting, handling, and storing fuel (both new and spent fuel) in the intermediate building and containments. The fuel handling system consists of equipment necessary to facilitate the periodic refueling of the reactor. The transfer of new fuel assemblies between the uncrating area and the new fuel storage vault is accomplished using the 10 ton auxiliary hook of the fuel handling area crane. The auxiliary hoist of the fuel handling platform transfers the new fuel from the storage vault to the transfer pool. The main hoist of the fuel handling platform handles the new or spent fuel assemblies over the spent fuel and transfers spent fuel to the cask loading pit. The fuel transfer system transports the new or spent fuel between the intermediate building and containment and the refueling platform handles both the new and spent fuel in containment.

The fuel handling area crane, the fuel handling platform and other components in the intermediate building are shared by both units. This sharing does not affect the fuel handling accident analysis and thus these shared components meet the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components."

The entire system is housed within the intermediate building and the containments which are seismic Category I, flood and tornado protected (refer to Sections 3.4.1 and 3.5.2 of this SER). The fuel handling area crane, the fuel handling platform (main ~~and~~ hoist), and the refueling platform are designed to seismic Category I requirements so that they will not fail in a manner which results in unacceptable consequences such as fuel damage or damage to safety related equipment. [The Final Safety Analysis Report does not state the seismic classification of the auxiliary hook of the fuel handling area crane and the auxiliary hoist of the fuel handling platform.] However, fuel handling systems are not required to function following an SSE. The new fuel inspection stand and the jib crane which is used for fuel preparation during refueling are designed to seismic Category I requirements. Thus, the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Bases" and 1.29, "Seismic Design Classification" relating to protection of safety-related equipment and spent fuel from the effects of earthquake, are satisfied, with the exception of the auxiliary components mentioned above.

The refueling platform is used to transport fuel and reactor components to and from pool storage and the reactor vessel. The fuel grapple hoist of the refueling platform has redundant load handling components so that no single component failure will result in a fuel bundle drop. Redundant interlocks and limit switches prevent accidental collision with pool walls. The design of fuel grapple in its fully raised position maintains adequate shielding by the water. The fuel handling platform is used to transport fuel within the intermediate building storage pool. Both the main fuel hoist and the auxiliary hoist have redundant load handling components so that no single component failure will result in a fuel bundle drop. Spent fuel will be handled with telescoping grapples designed to assure adequate shielding by the water.

Additionally, the height at which loads can be handled over the spent fuel by either the refueling platfrom or the fuel handling platform will be restricted by administrative procedures. Thus, the condition of a dropped object having a higher kinetic energy than the drop of a fuel assembly and its associated handling tool analyzed in Section 15 of the Final Safety Analysis Report is minimized.

The inclined fuel transfer system is used to transfer fuel, control rods and other small components between the containment and the fuel building pools.

The transfer operation is an automatic sequencing function with capability for manual override. Interlocks assure the correct sequencing of the transfer operation in the automatic or manual mode. Additional interlocks prevent the refueling platform and the fuel handling platform from moving in the transfer area during operations of the transfer system which would be adversely affected by the presence of either platform.

Based on the above, we conclude that the requirements of General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control" and 62, "Prevention of Criticality in Fuel Storage and Handling" and the guidelines of Regulatory Guide 1.13 with respect to prevention of unacceptable radioactivity releases and criticality accidents are satisfied.

Based on our review, we conclude that the fuel handling system is in conformance with the requirements of General Design Criteria 2, 5, 61, 62 and the guidelines of Regulatory Guides 1.13 and 1.29 with respect to protection of safety-related equipment and spent fuel from the effects of earthquakes and shared functions and prevention of unacceptable radioactivity releases and criticality accidents and is, therefore, acceptable, [pending the applicant's verification that the auxiliary hook of the fuel handling area crane and the auxiliary hoist of the fuel handling platform are designed to seismic Category I requirements.] We will report on the resolution of this matter in a supplement to this SER.

9.1.5 Overhead Heavy Load Handling

The overhead heavy load handling system consists of equipment necessary for the safe handling of the spent fuel cask and for safe disassembly, and reassembly of the reactor vessel head and internals during refueling operations. The containment polar crane is used for handling of heavy loads in containment and the fuel handling area crane is used for handling of heavy loads in the intermediate building.

The fuel handling area crane in the intermediate building is shared by both units. This sharing does not affect the fuel handling accident analysis and thus the fuel handling area crane meets the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components."

The entire system is housed within the intermediate building and containment which are seismic Category I, flood and tornado protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). Critical components of the fuel handling system are designed to seismic Category I requirements so that they will not fail in a manner which results in unacceptable consequences such as fuel damage or damage to safety-related equipment. However, fuel handling system components are not required to function following an SSE. The 125 ton fuel handling area crane is used for handling the 100 ton spent fuel shipping cask, and is designed to seismic Category I requirements. The containment polar crane is used to move the reactor vessel head, shroud head/separator, and dryer assembly, and is designed to seismic Category I requirements. Therefore, the design satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Bases" and 1.29 "Seismic Design Classifications."

The spent fuel cask pool is separated from the fuel storage pool by a canal with a seismic Category I gate. The spent fuel cask handling crane rails do not extend over any portion of the spent fuel storage pool, thereby preventing cask transportation over spent fuel. A dropped cask cannot, therefore, result in fuel damage. The crane coverage area does not include any area over safety-related equipment. Procedures and design limitation prevent the cask from being lifted more than 30 feet. Thus, we conclude that the requirement of General Design Criteria 4, "Environment and Missile Design Bases" and 61, "Fuel Storage and Handling and Radioactivity Control" and the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility" have been satisfied for handling of the spent fuel cask.

The applicant has not provided a load drop analysis for the containment polar crane. However, the applicant's response to NUREG-0612, "Control of Heavy Loads at Nuclear Power Plant," included the results of a load drop analysis for the polar crane." NUREG-0612 resolved Generic Task A-36 and provides guidelines for necessary changes to assure safe handling of heavy loads when the plant becomes operational and will include an evaluation of the load drop analysis. The applicant's response includes a commitment to implement the interim actions identified in Enclosure 2 attached to the December 22, 1980 generic letter prior to final implementation of the NUREG-0612 guidelines and prior to the receipt of their operating license. These interim measures deal with safe load paths, procedures, operator training and crane inspections, testing and maintenance. With the implementation of the interim actions required by NUREG-0612, we conclude that heavy load handling system is acceptable for licensing without completion of our evaluation of the applicant's response to NUREG-0612.

Based on our review, we conclude that the overhead handling systems are in conformance with the requirements of General Design Criteria 2, 4, and 61 as related to protection against natural phenomena, protection of safety-related equipment from the effects of internal missiles, and safe handling and storage of the fuel and the guidelines of Regulatory Guides 1.13 and 1.29 with respect to overhead crane interlocks and maintaining plant safety in a seismic event. We further conclude that implementation of the interim actions of NUREG-0612 prior to final implementation of NUREG-0612 guidelines and prior to receipt of the operating license provides reasonable assurance of safe handling of heavy loads until NUREG-0612 can be fully implemented. We conclude that the overhead handling systems are acceptable.

9.2 Water Systems

9.2.1 Station Service Water System (Emergency Service Water System)

The emergency service water system (ESWS) supplies cooling water to the plant from Lake Erie which serves as the ultimate heat sink as discussed in Section 9.2.5 of this SER. The ESWS operates during hot standby, cold shutdown, and accident conditions. Under these conditions, the ESWS provides cooling to the following essential plant components: the residual heat removal heat exchanger, the standby and high pressure core spray diesel generator heat exchanges, the emergency closed cooling system heat exchangers and the high pressure core spray pump room cooler. Additionally, the ESWS is capable of supplying water to flood containment for post-accident recovery, to provide emergency makeup to the fuel pool (from Unit 1 only), to de-ice the emergency service water pumphouse traveling screens and to provide emergency makeup to the emergency closed cooling system surge tanks. The ESWS discharges for the residual heat removal heat exchangers contain radiation monitors and the ESWS can be isolated on a high radiation alarm. A separate redundant ESWS is provided in each unit of the plant, thus the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components" are not applicable.

The ESWS consists of three independent piping loops labeled "A", "B", and "C" per unit, any two of which are required to assure safe shutdown. The "A" and "B" loops serve redundant RHR heat exchangers, diesel generator heat exchangers and emergency closed cooling system heat exchangers. The "C" loop serves only the HPCS diesel generator heat exchangers and support systems. Each loop is provided with a full capacity pump located in a separate cubicle in the emergency service water pump house. The ESWS pumps circulate water in an open cycle from Lake Erie through the components to be cooled and back to a different part of the Lake. Each division is powered from its associated diesel generator emergency bus.

The system is housed in seismic Category I, flood and tornado protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). Underground piping of the ESWS, which hydraulically connect these structures, is also protected from these natural phenomena. The system itself is designed to seismic Category I, Quality Group C requirements. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification" are satisfied.

The design of the ESWS described above assures that system function is not lost assuming a single active component failure coincident with a loss of offsite power. The applicant has provided sufficient information to assure that the ESWS is capable of transferring heat loads from safety-related components to the ultimate heat sink under all modes of operation. Therefore, we conclude that General Design Criterion 44, "Cooling Water" is satisfied.

The ESWS pumps are not normally operating. Their availability is assured by periodic functional tests and inspections as delineated in the plant Technical Specifications. The system design also incorporates provisions for accessibility to permit inservice inspection as required. The applicant has not addressed the plant provision to monitor possible flow blockage from sources such as Asiatic clams. Thus, we cannot conclude that the requirements of General Design Criteria 45, "Inspection of Cooling Water System" and 46, "Testing of Cooling Water System" are satisfied.

Based on the above, we conclude that the emergency service water system meets the requirements of General Design Criteria 2 and 44 with respect to the system's protection against natural pheonemena and capability for transferring the required heat loads, and the guidelines of Regulatory Guide 1.29, with respect to the system's seismic classification. However, we cannot conclude that the system meets the requirements of General Design Criteria 45 and 46 with respect to inservice inspection and functional testing. [The applicant is required to describe the plant provisions to monitor possible flow blockage in the emergency service water system resulting from sources such as Asiatic clams.] We will report on the resolution of this matter in a supplement to this SER.

9.2.2 Reactor Auxiliary Cooling Water System

Our review of the reactor auxiliary cooling water system included the emergency closed cooling system and the nuclear closed cooling system.

The emergency closed cooling (ECC) system provides cooling water to safety-related components only during hot standby, shutdown, and accident conditions. The system is a closed system with its heat exchangers cooled by the emergency service water system as discussed in Section 9.2.1 of this SER. When operating, the ECC system provides cooling to the following essential plants components: the low pressure core spray room cooler, the residual heat removal pump seals and room coolers, the reactor core isolation cooling room cooler, the control room chillers (Unit 1 ECC system only) and the fuel pool heat exchangers (Unit 2 ECC system only).

The ECC system consists of two independent piping loops any one of which is required to assure safe shutdown. Each loop consists of one pump, heat exchanger and surge tank with both loops sharing a nonsafety-related chemical addition tank. Each loop is powered from its associated diesel generator emergency bus. The components for each loop are located in separate areas of the control complex and intermediate building. The surge tanks are designed with a 40 day supply of water considering normal system leakage without makeup water. The emergency service water system provides makeup water by manual action from the control room.

A separate redundant ECC system is provided in each of the two units of the plant. The ECC system for Unit 1 serves the control room chillers during accident conditions. Cooling for the control room chillers is automatically transferred from the nuclear closed cooling system to the ECC system on a loss of coolant accident signal. The control room and control room chillers are common to both units of the plant. Cooling for the fuel pool heat exchangers can be manually transferred from the nuclear closed cooling system to the ECC system of Unit 2. Since a single active failure in the ECC systems will not prevent cooling of the common components and since the ECC systems for both units are designed for the heat loads of the common components, we conclude that the ECC systems satisfies the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components."

The ECC system is housed in seismic Category I, flood and tornado protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). The system is designed to seismic Category I, Quality Group C requirements except for the nonsafety-related chemical addition tank which is isolated by a normally closed seismic Category I, Quality Group C isolation valve. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification" are satisfied.

The design of the ECC system described above assures that the system function is not lost assuming a single active component failure coincident with the loss of offsite power. The applicant has provided sufficient information to assure that the ECC system is capable of transferring heat loads from safety-related component to the ultimate heat sink via the emergency service water system under all modes of operation. Therefore, we conclude that General Design Criterion 44, "Cooling Water" is satisfied.

The ECC system pumps are not normally operating. Their availability is assured by periodic functional tests and inspections as delineated in the plant Technical Specifications. The system design also incorporates provisions for accessibility to permit inservice inspection as required. Thus, we conclude that the requirements of General Design Criteria 45, "Inspection of Cooling Water System" and 46, "Testing of Cooling Water System" are satisfied.

The non-safety (non-seismic Category I, Quality Group D) nuclear closed cooling (NCC) system provides cooling water to auxiliary nuclear plant equipment during normal operations. The NCC system is a closed loop system which serves as a barrier to prevent leakage of reactor water into the service water system. The NCC provides cooling water to the following plant components: the recirculation pump motor bearing oil coolers and winding coolers, the recirculation pump seal coolers, the control rod drive pump, the reactor water cleanup pump, the drywell and containment sump heat exchangers, the drywell coolers, the instrument and service air compressors, the fuel pool heat exchangers, the control complex chillers and other components.

The NCC system is shared by both units and consists of three 50 percent pumps, three 50 percent heat exchangers and a surge tank. Two operating pumps and heat exchangers satisfy the maximum heat load requirement with the remaining pump and heat exchanger on standby. While the NCC system is operable from the emergency diesel generator buses, the system was evaluated and found to have no functions or heat loads necessary for achieving safe reactor shutdown condition or for accident prevention or accident mitigation. In the event that the system is inoperable, cooling water is available to the control room chillers and the fuel pool heat exchangers from the emergency closed cooling system as described above. Thus, the requirements of General Design Criteria 44, 45, and 46 are not applicable.

Redundant seismic Category I, Quality Group B isolation valves and piping are provided at the system's piping containment penetrations. The system's piping and valves associated with the control room chillers and fuel pool heat exchangers are seismic Category I Quality Group C. Protection from flooding of safety- related equipment resulting from failure of the system is discussed in Sections 3.4.1 and 9.3.3 of this SER. Failure of the system does not affect plant safe shutdown as described above, thus the requirement of General Design Criteria 2 and 5 and the guidelines of Regulatory Guide 1.29 are met.

Based on the above, we conclude that the emergency closed cooling system meets the requirements of General Design Criteria 2, 5, 44, 45 and 46 with respect to the system's protection against natural phenomena, shared functions, capability for transferring the required heat loads, inservice inspection and functional testing, and the guidelines of Regulatory Guide 1.29 with respect to the system's seismic classification and is, therefore, acceptable. Additionally, we conclude that the nuclear closed cooling system meets the requirements of General Design Criteria 2 and 5 with respect to the system's protection against natural phenomena and shared functions and the guidelines of Regulatory Guide 1.29 with respect to the system's seismic classification and is therefore, acceptable.

9.2.3 Demineralized Water Makeup System

The nonsafety-related (non-seismic Category I) demineralized water makeup system includes all components and piping associated with the system from the plant makeup water source, Lake Erie water, to the points of discharge to other systems and the waste discharge basin. The system has no safety-related function. Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Sections 3.4.1 and 9.3.3 of this SER. The system is capable of fulfilling the normal operating requirements of the facility for acceptable makeup water with the necessary component redundancy. Entry of potentially radioactive water into the system is precluded by assuring a greater pressure for demineralized makeup water than in the potentially radioactive sources to which it discharges. Alarmed instrumentation has been provided to prevent delivery of off-specification water to safety-related systems. Failure of the system will not affect plant safety as described above, thus the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena" and 5, "Sharing of Structures, Systems, and Component" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification" are met.

Based on the above, we conclude the system meets the requirements of General Design Criteria 2 and 5 with respect to the need for protection against natural phenomena and shared functions and meets the guidelines of Regulatory Guide 1.29 concerning the seismic classification and is, therefore, acceptable.

9.2.4 Potable and Sanitary Water Systems

The nonsafety-related (Quality Group D, non-seismic Category I) potable and sanitary water system provides water for human consumption and sanitary waste water treatment. The potable and sanitary water system is supplied from the Ohio Water Service Company via a water main extended onto the plant site. There are no cross-connections between the potable and sanitary water system and potentially radioactive systems. Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Sections 3.4.1 and 9.3.3 of this SER. Failure of this system does not affect plant safety as described above. Thus, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment" are met.

Based on our review, we conclude that the potable and sanitary water system meets the requirements of General Design Criterion 60 with respect to prevention of release of potentially radioactive water, and is, therefore, acceptable.

9.2.5 Ultimate Heat Sink

The ultimate heat sink, Lake Erie, provides makeup water to the cooling water system (natural draft cooling towers) by way of the service water system for normal plant operation. The emergency service water systems for both units operate during startup, shutdown, and emergency conditions by drawing water from the lake, cooling the plant, and returning the water to the lake. The lake has been shown to have a sufficiently high level to assure that it is always available to qualify as a single source of cooling water (refer to Section 2.4.1 of this SER).

Intake structures are located approximately one-half mile offshore and 13.3 ft below the low water datum level of the lake. A 10-ft diameter tunnel below the lake bed leads the water to the onshore pumphouse structures, the emergency service water pumphouse, and the service water pumphouse. Water is returned to the lake by way of a 10-ft diameter discharge tunnel below the lake bed to a discharge nozzle approximately 1500 ft offshore and 12 ft below the low water datum level of the lake.

The intake structures, discharge nozzle, inlet and discharge tunnels, emergency service water pumphouse, discharge tunnel entrance structure, and cross-tie between the emergency service water pumphouse and discharge tunnel entrance have all been designed to quality Group C, seismic Category I criteria and are shared by both units. The emergency service water pump house is a tornado protected structure. The other structures are either submerged or below grade so there is inherent protection from natural phenomena. We therefore conclude that the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.29, "Seismic Design Classification," and 1.27, "Ultimate Heat Sink for Nuclear Power Plants" are satisfied.

The applicant used GE Licensing Topical Report NEDO-10625, Class 1, March 1973 to determine the decay heat and conservative assumptions for the auxiliary heat loads. Because of the wide separation of the intake and discharge structures, recirculation of the water is prevented. The total heat rejected will have only a negligible thermal effect in the localized area and no thermal effects on the lake as a whole. Thus, sufficient water at a temperature below the design inlet temperature would be available for an indefinite period following a LOCA in one unit concurrent with loss of offsite power and shutdown of the other unit.

The tunnels and associated structures have been sized for simultaneous operation of the service water systems flow rate (70,630 gpm) and the emergency service water systems flow rate (45,400 gpm). In case of loss of the intake system for any reason, for any reason, the cross-tie permits use of the discharge system as a source of water to the emergency service water systems. Provisions have been made to discharge the emergency service water to the plant yard under these conditions. Thus, the requirements of General Design Criteria 44, "Cooling Water System," and 5, "Sharing of Structures, Systems, and Components," and the guidelines of Regulatory Guide 1.27, regarding the ability to maintain proper system temperature under all modes of operation are met.

Based on the above, we conclude that the ultimate heat sink meets the requirements of General Design Criteria 2, 5, and 44 with respect to protection against natural phenomena, missiles, pipe break effects shared system function, and heat dissipation capability, inservice inspection and functional testing, and the guidelines of Regulatory Guides 1.29 and 1.27 with respect to seismic classification and the capability to remove sufficient decay heat to maintain plant safety and, therefore, is acceptable.

9.2.6 Condensate and Refueling Water Storage Facilities

The nonsafety-related (quality Group D, non-seismic Category I) condensate storage and transfer system includes all components and piping associated with the system from the storage tanks to the points of connection or interfaces with other systems. The primary functions of the condensate storage system are to provide makeup to the main turbine cycle and to provide a dedicated water supply for the Reactor Core Isolation Cooling (RCIC) and the High Pressure Core Spray (HPCS) systems. The alternative water supply for the RCIC and HPCS systems for safe shutdown is the suppression pool. Additionally, the 500,000 gallon condensate storage tank (150,000 reserve for the above mentioned systems) provides demineralized water for the fuel pool cooling and cleanup system, the reactor water cleanup system, the refueling water system and other miscellaneous uses.

A separate condensate storage and transfer system is provided for each unit of the plant. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable.

The system was evaluated and found to have no functions necessary for achieving safe reactor shutdown conditions or for accident prevention or accident mitigation. Thus, the requirements of General Design Criteria 44, "Cooling

Water System", 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System" are not applicable.

The condensate storage tank is located within a seismic Category I dike which is designed to accommodate a total uncontrolled release from the tank. Protection from flooding for safety-related equipment resulting from failure of the system is discussed in Sections 3.4.1 and 9.3.3 of this SER. Seismic Category I, quality group B containment isolation valves are provided at the system's containment penetrations, and are located in seismic Category I, flood, tornado missile and environmentally protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). Additionally, the supply header for the RCIC and HPCS systems is seismic Category I, quality group B. Thus, the system meets the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification."

Based on our review, we conclude the system meets the requirements of General Design Criterion 2 with respect to the need for protection against natural phenomena and the guidelines of Regulatory Guide 1.29 concerning its seismic classification and is, therefore acceptable.

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

The compressed air systems include a safety-related instrument air system to supply the safety-related automatic depressurizing system (ADS) accumulators and a nonsafety-related (quality group D, non-seismic Category I) instrument and service air system to provide air for control purposes and for service purposes.

The safety-related instrument air system for each Unit consists of one reciprocating type air compressor purifier package, two air receiver tanks and associated piping to the ADS accumulators. These compressors continuously charge the air receiver tanks which in turn supply the ADS accumulators in order to provide assured long term ADS operability. The air is supplied to the

ADS accumulators from two physically separated air lines with each line supplying four ADS accumulators. Each line contains an air receiver tank and redundant check valves between the air receiver tank and the compressor to assure no backflow from the air receiver tank. Additionally, the safety-related instrument air system has a connection for recharging breathable air packs.

Each unit of the plant is provided with a separate safety-related instrument air system. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components" are not applicable.

The air compressor and piping upstream of the redundant isolation check valves are quality group D, non-seismic Category I. The air receivers, isolation check valves, accumulators and associated piping are quality group C, seismic Category I except for the containment penetration piping and isolation valves which are quality group B, seismic Category I. The system is located within seismic Category I, flood and tornado protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). The system is also protected from the effects of missiles and pipe breaks (refer to Section 3.5.1.1 and 3.5.1.2 and 3.6.1 of this SER). Thus, the system satisfies the requirements of General Design Criterion 2, "Design Basis for Protection Against Natural Pheonmena" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification."

A scheduled program of testing and inspection of the system will be provided to ensure operability of the system components and control systems. The compressor purifier package contains a purifier visual indicator which signals purifier cartridge replacement. For compliance with the requirements of General Design Criterion 1, "Quality Standards and Records" refer to Section 3.2 of this SER.

The nonsafety-related service air system for Units 1 and 2 each consists of one motor driven compressor with an integral aftercooler, an air intake filter silencer, a receiver tank and a piping system for distributing air throughout its associated plant unit. The service air compressor for each unit is capable of supplying air for both units and a cross-tie header between units is included in the distribution piping. During normal operation, one compressor supplies air for both units, with the other compressor on standby.

The nonsafety-related instrument air system for Units 1 and 2 each consists of one, oil-free compressor with after cooler, receiver tank, pre-filter, air dryer, and a piping system for distributing air throughout its associated plant unit. The service air system is interconnected with the instrument air system through an automatic control valve to provide a normal supply of air to the instrument air system. The instrument air system compressor serves as a backup. The service air system connection is upstream of the instrument air filters and dryers. Additionally, the instrument air system for Units 1 and 2 are cross-tied so that air can be supplied to either unit from any one of the units' instrument air systems.

The service and instrument air systems have no functions necessary for achieving safe reactor shutdown conditions and for accident prevention or mitigation. Instruments, controls and services required for safe shutdown of the plant such as the MSIV and ADS valves as discussed above are provided with seismic Category I passive air accumulators to assure their proper function in a loss of instrument air conditions. All other air-operated valves including the scram discharge inlet and outlet valves and other devices are designed for a fail-safe made upon loss of instrument air and do not require a continuous air supply under emergency or abnormal conditions. Additionally, all service and instrument air system containment penetrations are provided with redundant seismic Category I, quality group B isolation valves. Since a failure of the service and instrument air system will not prevent safe reactor shutdown, we conclude the requirements of General Design Criteria 2 and 5 and the guidelines of Regulatory Guide 1.29 are satisfied.

The instrument air system's design to supply clean, dry, oil free air is in accordance with ANSI Standard MC-11-1 (ISA-57.3), except that the maximum partial size is 5 micrometers instead of 3 micrometers. [The applicant didn't provide the basis for the exception to the standard. The applicant indicated that procedures would be developed to test the instrument air quality, however, the applicant did not indicate the test frequency or the testing locations. It is our position that the instrument air quality be tested at least yearly and that the test verify that the instrument air quality meets the above standard at the air filter-dryer discharge and at the end of each air line branch header, in order to assure continuous operation of the fail-safe valves throughout the system.]

A discussion of peroperational testing of the compressed air systems and compliance with Regulatory Guide 1.68.3, "Preoperational Testing of Instrument Air Systems," is contained in Section 14.0 of this SER.

Based on the above, we conclude that the safety-related air system meets the requirement of General Design Criterion 2 regarding protection against natural phenomena and the guidelines of Regulatory Guide 1.29 concerning seismic classification and the system is, therefore, acceptable. [However, we cannot conclude that the nonsafety-related service and instrument air system is acceptable until the applicant commits to testing of the instrument air quality at least yearly and that the test verifies the air quality at the air dryer-filter discharge and at the end of each air line branch header. Additionally, the applicant is required to provide the basis for the exception to the air quality standard concerning maximum ~~partial~~ particle size.] We will report on the resolution of this matter in a supplement to this SER.

(2) is capable of reducing occupational exposure to radiation by removing radioactive products from the pool water, and thus meets the requirements of Section 20.1(c) of 10 CFR Part 50, as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confines radioactive materials in the fuel pool water within the demineralizer and filters, and thus meets Regulatory Position C.2.f(2) of Regulatory Guide 8.8, as it relates to reducing the spread of contaminants from the source; and (4) removes suspended impurities from the pool water by filters, and thus meets Regulatory Position C.2.f(3) of Regulatory Guide 8.8, as it relates to removing crud through physical action.

On the basis of the above evaluation, we find that the spent fuel pool cleanup system meets GDC 61, Section 20.1(c) of 10 CFR Part 20 and the appropriate section of Regulatory Guide 8.8 and, therefore, is acceptable.

*9.3.2 Process Sampling System

I. INTRODUCTION

The process sampling system is designed to provide

* This evaluation is based upon applicant information through FSAR amendment 3 and by telecon and telex and is subject to confirmatory review of committed FSAR amendments.

representative samples, under controlled conditions, of plant process streams. Provisions for continuous monitoring of selected systems provide a means of analytical surveillance of system trends and performance during plant operations. Laboratory samples are taken to provide a) comprehensive analytical information on plant operations, b) a check on continuous monitoring instrumentation, and c) regular reports on critical plant systems to ensure safe and proper operation.

Components of the process sampling system which form part of the reactor coolant pressure boundary or containment isolation system are designed in accordance with seismic Category I requirements. Sample lines which form part of the containment isolation are provided with automatic fail-closed isolation valves both inside and outside containment.

II. EVALUATION AND FINDINGS

The system description and piping and instrumentation diagram were reviewed in accordance with SRP Section 9.3.2. The process sampling system includes piping, valves, heat exchangers, and other components associated with the system from the point of sample withdrawal from a fluid system up to the analyzing station, sampling station, or local sampling point. Our review included the provisions proposed to sample all principal fluid process streams associated with plant operation and the applicant's proposed design of these systems including the location of sampling points, as shown on piping and instrumentation diagrams.

We determined that the proposed process sampling system meets (1) the requirements of GDC 13 and 14 in Appendix A to 10 CFR Part 50 to monitor variables that can affect the reactor coolant pressure boundary and to assure a low probability of abnormal leakage, rapidly propagating filure, and gross rupture, respectively, by sampling the reactor coolant, and the condensate for chemical impurities that can affect the reactor coolant pressure boundary; (2) the requirements of GDC 13 and 16 in Appendix A to 10 CFR Part 50 to maintain the reactor core subcritical under cold conditions in the event that control rod system is inoperable, by sampling the standby liquid control system tank for boron concentration; and (3) the requirements of GDC 64 in Appendix A to 10 CFR Part 50, to monitor for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents, by sampling the reactor coolant, the main condenser evacuation system offgas, the sump inside containment, the drywell atmosphere, and the gaseous radwaste storage tank for radioactivity.

We further determined that the proposed process sampling system meets (a) the standards of ANSI N13.1-1969 for obtaining airborne radioactive samples; (b) the requirements of 10 CFR Part 20.1(c) and regulatory positions 2.d(2), 2.f(3), 2.f(8) and 2.i.(6) of Regulatory Guide 8.8 revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," to maintain radiation exposures to as low as is reasonably achievably, by providing (1) ventilation systems and gaseous radwaste treatment system to contain airborne radioactive materials; (2) liquid radwaste treatment system to contain radioactive material in fluids; (3) spent fuel pool cleanup system to remove radioactive contaminants in the spent fuel pool water; and (4) remotely operated containment isolation valves to limit reactor coolant loss in the event of rupture of a sampling line; (c) the requirements of GDC 60 to control the release of radioactive materials to the environment by providing isolation valves that will fail in the closed position; and (d) regulatory positions C.1, C.2, and C.3 of Regulatory Guide 1.26, revision 3, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive- Waste-Containing Components of Nuclear Power Plants," and C.1, C.3 and C.4 of Regulatory Guide 1.29, revision 3, "Seismic Design Classification," by designing the sampling lines and components of the process sampling system to conform to the classification of the system to which each sampling line and component is connected, and thus meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2.

On the basis of the above evaluation, we find that the proposed process sampling system meets the relevant requirements of 10 CFR Part 20, & 20.1(c), General Design Criteria 1, 2, 12, 14, 26, 60, 63, and 64 in Appendix A to 10 CFR Part 59, and Regulatory Guides 8.8, 1.26 and 1.29 and, therefore, is acceptable, pending confirmatory review of the amended FSAR.

*10.4.6 Condensate Filter Demineralizer System

I. INTRODUCTION

The condensate filter demineralizer system removes corrosion products, condenser inleakage impurities, and impurities present in the condensed steam. The system consits of the necessary piping, valves, appurtenances, and instrumentation to control the condensate impurity concentration during plant operation. Eight filters, in series with six deep bed demineralizers are provided to polish the condensate flow; six of the filters and five

* This evaluation is based upon applicant information through FSAR amendment 3 and by telecon and telex and is subject to confirmatory review of commited FSAR amendments.

9.3.3 Equipment and Floor Drainage System

The nonsafety-related (quality Group D, non-seismic Category I) equipment and floor drainage system includes all piping from equipment or floor drains to the sump, sump pumps, and piping necessary to carry potentially radioactive and non-potentially radioactive effluents through separate subsystems. Potentially radioactive drainage is collected in floor and equipment drain sumps in each building and discharged to the radwaste processing system. Drainage from non-potentially radioactive sources such as plumbing fixtures and roof drains are discharged to the sanitary waste treatment system and discharge basin, respectively. Thus the system design meets the pertinent requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

Containment penetrations for the equipment and floor drainage system are designed to seismic Category I and quality Group B requirements.

Our review considered those safety systems needed to provide safe plant shutdown and the physical location of those systems with regard to potential in-plant flooding. Because of their location at the lowest elevation in the auxiliary building, the ECCS equipment rooms which contain components required for safe plant shutdown were considered of particular importance with respect to provisions for prevention of water accumulation.

Each ECCS pump is located in an individual watertight room which contains a drainage pit to collect leakage from equipment within the room. The collected leakage then flows by gravity through an embedded line to the auxiliary building floor drain sump. Backflooding of the ECCS rooms is prevented by a manually operated normally closed valve in the line leading to the sump. Each drainage pit contains a level switch that operates an alarm in the control room to alert the operators to open the valve and that there is leakage in the room. Thus a single non-seismic valve without position indication is relied on to prevent backflooding of safety-related equipment and a single level switch is relied on to indicate leakage from safety-related equipment. We therefore cannot conclude that the system meets the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," or 4, "Environmental and Missile Design Bases," concerning protection against internal plant flooding as a result of postulated piping failures; it is therefore unacceptable.

Based on our review, we conclude that the system meets the requirements of General Design Criterion 60 with respect to protection against releases of radioactive material to the environment. [However, we cannot conclude that the system meets the requirements of General Design Criteria 2 and 4. We require resolution of the following items:

1. In Section 9.3.3.2.1, the FSAR indicates that a level switch is installed in each drainage pit in each of the ECCS pump rooms to alarm in the control room in case of leakage from the safety-related equipment in the room. Figure 11.2-7 sheets 3 and 4, show one level switch per room.

Verify that there are redundant safety grade switches or show how a single non-safety grade switch is qualified to perform this function.

2. In Section 9.3.3.3 you state that flooding of the ECCS rooms by backflow through the floor drains is prevented by a normally closed manually operated valve in the drain line. The PI&Ds do not show that the valves have position indicating transmitters.

   a. Describe the methods that will be used to assure that these valves will be closed during normal plant operation.

   b. The P&IDs and text indicate that the drainage system and valves are of non-seismic Category I design. Explain how backflooding would be prevented following an SSE that would cause failure of non-seismic piping systems.]R We wil report on the resolution of this matter in a supplement to this SER.

9.3.5 Standby Liquid Control System

The standby liquid control system (SLCS) is a reactivity control system, its purpose being to inject sodium pentaborate into the primary system to provide an independent means for shutting down the reactor should the normal reactivity control system become inoperable; thus, satisfy the requirements of General Design Criterion 26, "Reactivity Control System Redundancy and Capability." (Refer to Section 4.6 of this SER for the discussion of reactivity control.) The system consists of a storage tank, a test tank, two positive displacement pumps, two explosive-actuated valves, and associated local valves, piping and controls located within the containment. An electrical resistance heating system maintains the solution storage tank and pump suction lines between 75 and 85 degrees Fahrenheit to prevent precipitation of the sodium pentaborate from solution during storage. High and low liquid level and temperature are alarmed in the control room. The two explosive-actuated valves provide assurance that they will be opened when needed and due to the design of the valve it is ensured that boron will not leak into the reactor even during SLCS pump testing. The two parallel pumps draw the solution from the storage tank via two suction lines and discharge it into the reactor vessel via a common injection line. The discharge from each pump is provided with a check-valve, a crossover line, and an explosive valve. Each pump and its associated valves are powered from separate emergency ac power supplies. They are arranged such that failure of a single pump or explosive valve will not prevent adequate amounts of sodium pentaborate solution from entering the reactor vessel to accomplish shutdown.

System initiation is accomplished by manual actuation of either of two key-locked switches on the control room panel. Changing either switch status to "run" starts an injection pump, actuates an explosive valve, opens a tank outlet valve and closes reactor cleanup system isolation valves to prevent loss or dilution of boron. Should the instrumentation provided indicate that the solution is not entering the reactor vessel, the operator can turn the other key-operated switch to the "run" position to actuate the alternate equipment.

The SLCS is located in a compartment within the seismic Category I, flood- and tornado-protected containment building. All portions of the SLCS necessary for injection of sodium pentaborate into the reactor are seismic Category I, Quality Group B, or Quality Group A if they are part of the reactor coolant pressure boundary. Thus the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification" are met.

The SLCS is designed to function in the expected environmental conditions. The containment compartment in which the system is located provides protection against external or internally generated missiles. The SLCS is separated from non-seismic system components and from the effects of breaks in other high- and moderate-energy piping systems (refer to Sections 3.5.1.2 and 3.6.1 of this SER).

The SLCS is redundant such that no single active failure will compromise its functional capability. The injection portion of the system can be functionally tested by injecting demineralized water from a test tank into the reactor. Thus the requirements of General Design Criterion 27, are satisfied.

Based on the above, we conclude that the standby liquid control system is in conformance with the requirements of General Design Criteria 2, 26, and 27 as they relate to protection against natural phenomena, system function and system redundancy and testability, and the guidelines of Regulatory Guide 1.29 relating to the system's seismic classification and is, therefore, acceptable.

9.4.1 Control Room Area Ventilation System

The control complex is served by five separate heating, ventilating and air conditioning (HVAC) systems. These systems are as follows: (1) the control room HVAC system, (2) the motor control center, switchgear and miscellaneous electrical equipment areas HVAC system, (3) the battery room exhaust system, (4) the controlled access and miscellaneous equipment areas HVAC system, and (5) the computer rooms HVAC system. The control room HVAC system covers the control rooms and the extended term habitability area of the control building (refer to Section 6.4 of this SER for further discussion of control room habitability.)

The motor control center, switchgear and miscellaneous electrical equipment areas HVAC system is shared between Units 1 and 2. The system consists of two fully redundant sets of chilled water cooling coils, filters, fans, ductwork associated with each set and redundant isolation dampers. Non-redundant ductwork serves the areas to be cooled. Both sets are powered from ESF buse so that emergency power is available from the diesel-generators if offsite power is lost. Chilled water for the cooling coils is supplied by a safety-related chilled water system described in Section 9.4.5 of this SER. An indication of low flow from the operating set automatically starts the redundant set, repositions the pertinent dampers, and sounds an alarm in the control room. Thus, the design conforms to the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components."

The system is designed to remove the heat generated in the areas served and maintain the environmental conditions within the limitations of the equipment involved during all operating modes including LOCA conditions. All essential portions of the system are located in the control building which is a seismic Category I, flood- and tornado-protected structure. The system itself is designed to seismic-Category-I, Quality-Group C requirements. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," are met.

The system is physically separated from high-energy systems and is thus protected from the effects of postulated pipe failures in high-energy systems. Each equipment set is located in a separate missile-protected room provided with suitable drainage for protection against flooding due to moderate-energy piping system failures and failures in non-seismic water systems. The system air intakes are provided with tornado missile barriers. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases" are satisfied. The equipment in the motor control center, switchgear and miscellaneous electrical areas are not required for control of releases of radioactive materials to the environment, and thus the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment" and the guidelines of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" and 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" are not applicable.

Purging of smoke, carbon dioxide, or other contaminants from the areas served by the motor control center, switchgear and miscellaneous electrical equipment areas HVAC system may be accomplished by utilizing the exhaust capabilities of the Battery Room Exhaust System.

The battery room exhaust system is shared between Units 1 and 2 and consists of two fully redundant sets of exhaust blowers, exhaust ductwork associated with each set and isolation dampers. Non-redundant ductwork serves the areas to be purged. The motor control center, switchgear and miscellaneous electrical equipment areas HVAC system supplies the purge air. Both sets are powered from separate ESF buses. An indication of low flow from the operating set automatically starts the redundant set, repositions the pertinent dampers and sounds an alarm in the control room. Thus, the design conforms to the requirements of General Design Criterion 5.

The system is designed to remove combustible gas from the battery rooms and exhaust the control room lavatories, conference room, and kitchen during all operating modes including LOCA conditions. The sets are located in the same separate protected rooms which contain the motor control center, switchgear and miscellaneous electrical equipment areas HVAC system, are designed to seismic-Category-I, Quality-Group-C requirements, and have missile-protected exhaust louvers. Thus, the requirements of General Design Criteria 2, and 4 and the guidelines of Regulatory Guide 1.29 are satisfied.

The equipment in the battery rooms is not required for control of releases of radioactive material to the environment, and thus, the requirements of General Design Criterion 60 and the guidelines of Regulatory Guides 1.52 and 1.140 are not applicable.

The control room HVAC system is shared between Units 1 and 2. The system consists of two fully redundant sets of chilled water cooling coils, filters, fans, electric heaters, humidifiers, ductwork associated with each set, isolation dampers, and tornado-missile-protected inlet and exhaust louvers. Each set has a parallel emergency recirculation system containing fans, heaters, ductwork and dampers as well as demisters, HEPA filters, and activated charcoal beds for the removal of radioactivity and noxious gases. Non-redundant ductwork serves the areas to be cooled. Power to the system components for each set is supplied from redundant ESF buses. Chilled water to the cooling coils is provided by a safety-related chilled water system described in Section 9.4.5 of this SER. An indication of low flow from the operating set automatically starts the redundant set, repositions the pertinent dampers and sounds an alarm in the control room. [The control room HVAC system's controls, alarms, readout instruments, etc., are located in the control room; however, a description of the method of operation of this equipment has not been provided. Therefore, we cannot conclude that the requirements of General Design Criterion 5 have been satisfied.]

The control room HVAC system is designed to maintain the control room within the environmental limits required for operation of plant controls and for uninterrupted safe occupancy of required manned areas during all operating modes including LOCA conditions. The system is designed to maintain the control room under positive pressure. The applicant did not state whether radiation and chlorine detectors are provided in the intake ducts to automatically isolate the system and turn on the emergency recirculating system. Thus, we cannot conclude that the requirements of General Design Criteria 19, "Control Room," and 60 and the guidelines of Regulatory Guides 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,"1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," 1.52 and 1.140 are satisfied. Also, the applicant has not stated that all parts of the system are designed to seismic-Category-I, Quality-Group-C requirements. Thus, we cannot conclude that the requirements of General Design Criterion 2 and the guidelines of Regulatory Guide 1.29 are met.

The control room HVAC system trains are located in the same separate protected rooms which contain the motor control center, switchgear and miscellaneous electrical equipment areas HVAC system equipment. The system air intakes are provided with tornado missile barriers. The system incorporates provisions for the purging of smoke or other contaminants with no recirculation by bringing in fresh outside air and exhausting the contaminated air to the outside. Thus, the requirements of General Design Criteria 4 are satisfied.

The controlled access and miscellaneous equipment areas HVAC system is nonsafety-related (Quality Group D, non-seismic Category I) and is designed to maintain an acceptable environment in the controlled access and miscellaneous equipment areas of the control building. Of these areas, only the emergency closed cooling system rooms contain essential equipment. Under emergency or loss-of-offsite-power conditions, these rooms are cooled with an independent safety-related system described in Section 9.4.5 of this SER. Ventilation to the controlled access and miscellaneous equipment areas is not required for safe plant shutdown or under accident conditions; thus, failure of the system will not compromise plant safety. This cooling system is located in the seismic-Category-I, flood- and tornado-protected control building and is separated from the effects of missiles, and pipe failures in high energy systems. Thus, the requirements of General Design Criteria 2, 4, 5 and 19 concerning protection against natural phenomena, missiles and environmental effects, shared systems, and ability to maintain a proper personnel and equipment operating environment, and the guidelines of Regulatory Guides 1.29 relating to the system's seismic classification are satisfied.

The equipment in the controlled access and miscellaneous equipment areas are not required for control of releases of radioactive materials to the environment, and thus the requirements of General Design Criterion 60 and the guidelines of Regulatory Guides 1.52 and 1.140 are not applicable.

The computer rooms HVAC system is nonsafety-related (Quality Group D, non-seismic Category I) and is designed to maintain an acceptable environment in the Unit 1 and Unit 2 computer rooms and adjacent cable spreading rooms. None of the areas contains essential equipment. Ventilation to the computer rooms and adjacent cable spreading rooms is not required for safe plant shutdown or under accident conditions; thus, the failure of the computer rooms HVAC system will not compromise plant safety. This HVAC system is located in the seismic-Category-I, flood- and tornado-protected control building and is separated from the effects of missiles, and pipe failures in high-energy systems. Thus, the requirements of General Design Criteria 2, 4, 5 and 19 concerning protection against natural phenomena, missiles and environmental effects, shared systems, and the ability to maintain proper computer room equipment and personnel operating environment, and the guidelines of Regulatory Guides 1.29 relating to the system's seismic classification are satisfied.

The equipment in the computer rooms is not required for control of releases of radioactive materials to the environment, and thus the requirements of General Design Criterion 60 and the guidelines of Regulatory Guides 1.52 and 1.140 are not applicable.

Based on the above, we conclude that the motor control center, switchgear and miscellaneous electrical equipment areas HVAC system, the battery room exhaust system, the controlled access and miscellaneous equipment areas HVAC system, and the computer rooms HVAC system are in conformance with the requirements of General Design Criteria 2, 4, 5 and 19 relevant to protection against seismic events, floods, tornadoes, missiles, environmental effects, shared systems function, and ability to maintain an acceptable environment for humans and equipment, and the guideline of Regulatory Guides 1.29 relating to the system's seismic classifications and are, therefore, acceptable. [However, there are possible deficiencies in the design of the control room HVAC system and we cannot determine if it conforms to the requirements of General Design Criteria 19 and 60 and the guidelines of Regulatory Guidelines 1.52 and 1.140, relating to personnel radiation protection, and to the requirements of General Design Criterion 2 and the guidelines of Regulatory Guides 1.29, 1.78, and 1.95 relating to the system's seismic classification and protection against hazardous materials and chlorine gas release and to the requirements of General Design Criterion 5 relating to shared functions; it is, therefore, unacceptable. We require the resolution of the following items:

(1) We note that parts of Table 3.2-1, Classification of Systems, Components and Structures, are incomplete. In Section XXXV, Heating, Cooling, and Ventilation Systems, expand items 11, 12, 17, 18, 20, 24, 25, 26 and 27 to show (a) fans and motors, (b) cooling coils, (c) filters, (d) humidifiers, (e) charcoal filter housings, (f) ductwork and dampers, (g) valves with safety isolation function, (h) electrical and unit heaters, (i) electrical modules with safety function, (j) cable with safety function, as applicable to a given system. Expand items 31 and 32 to show all equipment such as motors, pumps, heat exchangers, piping, valves, and electrical modules and cable with a safety function.

(2) Table 6.4-4, Control Room Emergency Filter System, Single Failure Analysis covers the emergency filtration subsystem only. Provide a single failure analysis for the control room HVAC system.

(3) In Section 6.4.2.3, Leak Tightness, the FSAR states that the inlet ducts contain two isolation dampers in series. These are shown in Figure 6.4-1, the P & ID for the system. The FSAR further states that the exhaust isolation dampers are similar in arrangement to the inlet dampers. However, the P & ID shows only one damper per exhaust duct. Confirm that

there are two dampers in series in each exhaust duct and correct Figure 6.4-1 accordingly.

(4) In Section 2.2.3.1.2.1, the FSAR states that chlorine and ethylene oxide detectors will automatically initiate isolation of the control room upon detection of their respective gases. These detectors are not shown in Figure 6.4-1. Radiation detectors are also not shown on the inlet ducts. Revise Figure 6.4-1 to show redundant chlorine, ethylene oxide, and radiation detectors in each of the inlet ducts that will automatically initiate isolation of the control room upon detection.

(5) In Section 6.4, the FSAR states that the control room HVAC system controls, alarms, readout instruments, etc., are located in the control room. Since there are two control rooms and, presumably, two sets of operators--one for each unit, describe where the panel(s) for this system is located and discuss the method of operation of this shared system under all modes of station operation.

(6) Figure 1.2-6 of the FSAR shows the six emergency diesel exhaust silencers on the roof of the diesel-generator building at an elevation of approximately 650 ft and approximately 75 ft west of the control building wall.

Figure 1.2-9 shows two air intakes on the west wall of the control building at the approximate elevation of 680 ft. It also shows air intakes on the north and south walls of the control building at the same elevation. Describe the systems serviced by these air intakes. In the event of a west wind simultaneous with the diesel operation, it seems possible that poisonous diesel exhaust fumes could be drawn into the control building. Show either that this cannot occur or that would not pose a threat to the safe conduct of operations of the station.]

We will report on the resolution of this matter in a supplement to this SER.

9.4.2 Spent Fuel Pool Area Ventilation System

The spent fuel pool area ventilation system is designed to maintain a suitable environment for equipment operation and to limit potential radioactive release to the atmosphere during normal operation and postulated fuel handling accident conditions. The system is not required for safe shutdown of the plant in the event of a LOCA but is required to mitigate the consequences of a fuel handling accident. The system is classified as seismic Category I, Quality Group C.

The system serves the fuel pool area, the control rod drive pump areas, the fuel pool cooling and cleanup equipment rooms, the hot I&C repair shop, and the intermediate building sump pump room.

The system consists of a tornado missile protected air intake, roughing filters and heating coil, two 100% capacity supply fans, supply ductwork, exhaust ductwork, three 50% capacity filtration exhaust units (each with an electric heating coil, roughing filter, HEPA filters, charcoal adsorber and exhaust fan), and necessary dampers. The system exhausts to the Unit 1 exhaust stack. Power is supplied from the Unit 1 emergency diesel generator buses. The equipment must be restarted manually following a LOCA or loss of offsite power. The system has sufficient redundancy to satisfy the single active failure criterion.

During normal and refueling operations, the system maintains a slightly negative pressure in the fuel handling area to ensure that any airborne radioactivity is collected by the system. This is accomplished by continously operating the system. The airflow pattern is from potentially low radioactivity areas to potentially higher radioactivity areas. Slightly more air is exhausted than is supplied, thereby preventing short circuiting of air flow and assuring that no ambient air escapes the fuel handling area prior to being processed by the charcoal filtration system. Radiation detectors located in the ducts upstream of the charcoal filters will automatically shut down the supply fan on indication of high radioactivity and alarm in the control room. The exhaust system remains in operation to continue exhausting contaminated air through charcoal filters, thus precluding uncontrolled release of radioactivity to the outside and spread of radioactivity within the building.

We conclude that the above provisions adequately meet the pertinent requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," and 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," for preventing release of radioactive contaminants to the environment. Our review has determined that the system is capable of fulfilling the requirements of the facility for providing a fuel handling area environment with controlled temperature to ensure the comfort and safety of personnel and the integrity of fuel handling equipment during normal operation and during fuel handling operations.

The system is seismic Category I, Quality Group C and is located in the intermediate building which is seismic Category I, flood- and tornado-protected, thereby satisfying the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification." There are no high- or moderate-energy systems located near the system and adequate protection against internally and externally generated missiles is provided by separated equipment locations. (Refer to Section 3.5.1.1 and 3.6.1 of this SER).

The system serves the shared spent fuel storage facility. It has sufficient redundancy and separation to perform its cooling and filtration function so that no single active failure coincident with loss of offsite power will cause failure of the system. It therefore meets the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components."

Based on the above, we conclude that the spent fuel pool area ventilation system is in conformance with the requirements of General Design Criteria 2, 4, 5, 60 and 61 as they relate to protection against natural phenomena, sharing of systems, radioactive releases, fuel storage radioactivity control and the guidelines of Regulatory Guides 1.13, 1.29, and 1.52 relating to protection against radioactive releases, seismic classification, and system design for radioactivity control and is, therefore, acceptable.

9.4.3 Auxiliary and Radwaste Area Ventilation System

The auxiliary and radwaste area ventilation system serves the radwaste building, the auxiliary building, and the intermediate building except for the fuel handling area. Three separate systems serve these areas. The radwaste building (shared by Units 1 and 2) ventilation system provides ventilation for the radwaste building, identical auxiliary building ventilation systems provide ventilation for each of the auxiliary buildings (for Units 1 and 2), and the intermediate building (shared by Units 1 and 2) ventilation system provides ventilation for those areas of the intermediate building not served by the spent fuel pool area ventilation system.

The radwaste building ventilation system is classified as nonsafety-related (Quality Group D, non-seismic Category I). The ventilation system is capable of fulfilling the requirements of the facility for providing an environment with controlled temperature and air flow to ensure both the comfort and safety of plant personnel and the integrity of the non-essential equipment and components served. Equipment and instrumentation have been provided with suitable redundancy to ensure normal operation and to prevent release of radioactivity to the environment and thus the system is acceptable for its designed task. Failure of the system does not compromise the operation of any essential systems and does not affect the capability to safely shut down the plant or result in unacceptable release of radioactivity; thus the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," and Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are met.

The auxiliary building ventilation system is classified as nonsafety-related (Quality Group D, non-seismic Category I). The system has been designed to provide an environment with controlled temperature and air flow patterns to ensure the comfort and safety of personnel and the integrity of safety-related and nonsafety-related auxiliary building equipment on a normal operating basis and is acceptable. Environmental control for safety-related equipment during accident conditions is reviewed under the engineered safety feature ventilation system and is discussed in Section 9.4.5 of this SER. Failure of the system does not compromise the operation of any essential systems and does not affect the capability to safely shut down the plant or result in unacceptable release of radioactivity. Thus the requirements of General Design Criterion 2, and the guidelines of Regulatory Guide 1.29 with respect to protection against natural phenomena, and the requirements of General Design Criterion 60 and the guidelines of Regulatory Guide 1.140 with respect to the capability to provide normal ventilation exhaust and air filtration are satisfied.

The intermediate building ventilation system is classified as nonsafety-related (Quality Group D, nonseismic Category I). The system has been designed to provide an environment with controlled temperature and air flow patterns to ensure the comfort and safety of personnel and the integrity of safety-related and nonsafety-related intermediate building equipment on a normal operating basis. However, the system provides the environmental control for the following safety-related equipment:

1. Spent fuel pool area ventilation system exhaust trains (three separate rooms).
2. Annulus exhaust gas treatment system trains (four separate rooms, two per unit).
3. The hydrogen recombiner equipment areas (four separate areas, two per unit).
4. Supplies make-up air to the fuel pool cooling circulating pump room (exhaust from this room is provided by the safety related spent fuel pool area ventilation system).

In case of accident, single active failure, or loss of offsite power, it is not known how the environmental control of these areas is provided. Also, it is not known if the outside air inlet for the system is provided with tornado missile barriers to prevent entry of missiles to the intermediate building.

Based on the above, we conclude that the radwaste building ventilation system and auxiliary building ventilation system are in conformance with the requirements of General Design Criteria 2 and 60 as related to protection against natural phenomena and control radioactive release and the guidelines of Regulatory Guides 1.29 and 1.140 with respect to seismic classification and normal ventilation exhaust and air filtration and are, therefore, acceptable. However, we cannot conclude that the intermediate building ventilation system meets the requirements of General Design Criteria 2 and 60 and the guidelines of Regulatory Guides 1.29 and 1.40. [The applicant is required to provide the following:

1. Describe the means provided to assure that the temperature in the room housing the spent fuel pool cooling pumps, the rooms housing the hydrogen recombiner equipment, the rooms housing the annulus exhaust gas treatment system trains, and the rooms housing the spent fuel pool area ventilation system exhaust trains can be maintained at acceptable levels for equipment operation under accident and emergency conditions when the normal intermediate building ventilation system is not operating.

2. Describe the means provided to prevent tornado missiles from entering the intermediate building through the intermediate building ventilation system air inlet.]

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

9.4.4 Turbine Area Ventilation System

Two identical turbine building ventilation systems (one for each unit) provide the turbine building air flow requirements and are classified as nonsafety-related (Quality Group D, non-seismic Category I). The ventilation systems are capable of adequately maintaining an acceptable environment for personnel and non-essential equipment served during normal plant operation. Failure of a system does not compromise the operation of any essential systems and does not affect the capability to safely shut down the plant or result in unacceptable release of radioactivity; thus the requirements of General Design Criterion 2," Design Bases for Protection Against Natural Phenomena" and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," are met. Conversely, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment and the guidelines of Regulatory Guide

1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are not applicable.

Based on our review, we conclude that the turbine building ventilation systems meet the requirements of General Design Criterion 2 with respect to the need for protection against natural phanomena and the guidelines of Regulatory Guide 1.29 concerning its seismic classification and are, therefore acceptable.

9.4.5 Engineered Safety Feature Ventilation System

The engineered safety feature ventilation system provides cooling for equipment in the emergency service water pump house, the diesel-generator building, the emergency core cooling system pump rooms and the emergency closed cooling pump area and also includes the control complex chilled water system. The equipment in these areas are not required for control of releases of radioactive materials to the environment, and thus the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment" and the guidelines of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" and 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" are not applicable.

Emergency Service Water Pump House Ventilation System

The emergency service water pumps for Units 1 and 2 are located in a shared emergency service water pump house. A separate ventilation system is provided for each of the two unit's areas in the pump house. Each emergency service water pump house ventilation system consists of two 100%-capacity supply fans, two 100%-capacity relief louvers, supply and return ductwork, dampers, and controls to assure a proper ambient environment under all operating modes. Nonseismic Category I electric unit heaters maintain space temperatures in winter to prevent freezing during those times the emergency service water pumps are not in use. Each train is powered from the same emergency bus as the pump it serves and is automatically started when its corresponding pump is started. The above design assures system function in the event of a single failure.

The systems are housed in the emergency service water pump house which is seismic Category I, flood- and tornado-protected, and the systems are themselves designed to seismic Category I, Quality Group C requirements, thereby satisfying the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification." The pump house is designed against the effects of tornado missiles and is separated from high-energy piping systems and internally generated missiles, thereby satisfying the requirements of General Design Criterion 4, "Environmental and Missile Design Bases." There is no sharing of systems; thus the requirements of Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

Diesel Generator Building Ventilation System

The diesel generator building is shared by Units 1 and 2. Each diesel generator is located in a separate room within the building (there are six diesel--three per unit). The diesel generator building ventilation system consists of six independent subsystems, one for each diesel generator room, to assure adequate air flow in the event of a single failure. Each subsystem has two redundant full-capacity ventilation trains of ductwork, fan, dampers, and controls to maintain room cooling when the diesels are operating, and non-seismic Category I electric fan coil units to maintain room temperature in the winter when the diesel is not running. Each subsystem is powered from its respective emergency bus, and both trains are automatically started when their respective diesel is started.

The system is designed to seismic Category I, Quality Group C requirements and is housed in the seismic Category I, flood- and tornado-protected diesel-generator building, thereby satisfying the requirements of General Design Criterion 2 and the guidelines of Regulatory Guide 1.29. The inlet and outlet louvers are tornado-missile protected as is the diesel-generator building. The system is separated from high-energy piping systems and internally generated missiles, thereby satisfying the requirements of General Design Criterion 4. There is no sharing of systems; thus the requirements of General Design Criterion 5 are not applicable. The inlet louvers are approximately thirty feet above grade thereby meeting the guidance of item 2, subsection A, of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," and therefore the pertinent requirements of General Design Criterion 17, "Electric Power Systems," relating to the protection of essential electrical components from failure due to the accumulation of dust and particulate material are satisfied.

ECCS Pump Rooms Cooling Systems

The emergency core cooling (ECCS) system pump rooms are served by individual fan coil cooling units--one per ECCS pump room, thus assuring adequate air in the event of a single failure. The system serves RHR pump rooms A, B, and C, and the HPCS pump room, LPCS pump room, and RCIC pump room. Each fan coil unit is powered from the same emergency bus as the equipment it serves. The pump room units are automatically started when their respective pumps start. Cooling water to all the units except that in the HPCS pump room is supplied by the safety-related emergency closed cooling water system. The HPCS pump room unit is supplied by the safety-related emergency service water system. Each fan coil unit is designed to seismic Category I, Quality Group C requirements and is housed in the seismic Category I, flood- and tornado-protected auxiliary building, thereby satisfying the requirements of General Design Criterion 2 and the guidelines of Regulatory Guide 1.29. Each unit is separated from high-energy piping systems and internally generated missiles by locating it in the individual protected ECCS pump room, and is protected from tornado missiles by the tornado-missile-protected auxiliary building, thereby meeting the requirements of General Design Criterion 4. The units are not shared; thus the requirements of General Design Criterion 5 are not applicable. The above-described design for cooling of the HPCS and RCIC pump rooms also meets the requirements of TMI-2 Task Action Plan, NUREG-0737, Item II.K.3.24, "Confirmation of Adequacy of Space Cooling for High-Pressure Coolant Injection and Reactor Core Isolation Cooling Systems", by providing an emergency power supply from the diesel generators.

Emergency Closed Cooling Pump Area Cooling System

The emergency closed cooling pump areas are located in the shared control complex building. One area contains the Unit 1 and Unit 2 emergency closed cooling "A" pumps and heat exchangers and one of the control complex chillers and chilled water pumps as discussed below. The other area, separated from the first area, contains the Unit 1 and Unit 2 emergency closed cooling "B" pumps and heat exchangers and the other control complex chiller and chilled water pump. The cooling system consists of two 100%-capacity (100% capacity means the capacity to cool both areas) air handling units containing roughing filters and chilled water coils, supply distribution ducts and dampers. The dissipated heat is removed by cooling water from the safety-related control complex chilled water system. Each system is powered from the same emergency bus as the Unit 1 equipment it serves. Operation of this system is initiated automatically upon receipt of a start signal from the associated emergency closed cooling pump circuitry. The systems are housed in the control building which is seismic Category I, flood- and tornado-protected, and the systems themselves are designed to seismic Category I, Quality Group C requirements, thereby satisfying the requirements of General Design Criterion 2 and the guidelines of Regulatory Guide 1.29. The system is separated from high-energy piping systems and internally generated missiles, thereby satisfying the requirements of General Design Criterion 4. Shared cooling of Unit 1 and Unit 2 safety-related equipment is the result of this design. Because of equipment redundancy and separation--both in the equipment to be cooled and in the cooling system--such sharing does not significantly impair their ability to perform their safety functions, and thus the requirements of General Design Criterion 5 are met.

Control Complex Chilled Water System

The control complex chilled water system (CCCWS) provides mechanically chilled cooling water to the cooling coils of the following redundant safety-related control complex air handling units (refer to Section 9.4.1 of this SER):

Control room

Motor control center, switchgear and miscellaneous areas
Emergency closed cooling pump area.

It also provides cooling water to the cooling coils of the following redundant nonsafety-related and nonseismic air handling units:

Controlled access and miscellaneous equipment areas Computer room.

Motor-operated isolation valves will automatically isolate these coils if a failure occurs. The CCCWS consists of three 100%-capacity water chillers and three 100%-capacity circulating pumps connected to two redundant chilled water piping systems, one of which is normally operating with the other on manual standby. Loop A can be connected to chiller A or C, and loop B can be connected to chiller B or C. One chiller and circulating pump is connected to the Unit 1 division 1 emergency bus. Another chiller and pump is connected to the Unit 1 division 2 emergency bus. The third chiller and pump is connected to the Unit 2 division 1 emergency bus. This provides redundant availability of power during all periods of normal or emergency operation of the plant. Heat rejected by the mechanical chillers is absorbed by the nonsafety-related nuclear closed cooling water system during normal plant operation. During emergency conditions, the safety-related emergency closed cooling water system is used.

The system is housed in the control building which is seismic Category I flood- and tornado-protected, and all the components, piping and valves are designed to seismic Category I, Quality Group C requirements, thereby satisfying the requirements of General Design Criterion 2 and the guidelines of Regulatory Guide 1.29. The system is separated from high-energy piping systems and internally generated missiles, thereby satisfying the requirements of General Design Criterion 4. Shared cooling of Unit 1 and Unit 2 safety-related equipment is the result of this design. Because of equipment redundancy and separation--both in the equipment to be cooled and in the cooling system--such sharing does not significantly impair their ability to perform their safety functions, and thus the requirements of General Design Criterion 5 are met.

Based on the above, we conclude that the engineered safety feature ventilation system is in conformance with the requirements of General Design Criteria 2, 4, 5 and 17 (diesel-generator room ventilation system only) as they relate to protection against natural phenomena and assurance of proper operational environment for essential equipment including the diesel generators and the guidelines of Regulatory Guide 1.29 relating to the system's seismic classification. We also conclude that TMI-2 Task Action Plan, NUREG-0737, Item II.K.3.24 is satisfied.

[However, no mention was made of the ventilation requirements of the hydrogen recombiner equipment. Until we are assured either that this safety-related equipment needs no vetilation during post-accident conditions or is provided with safety grade ventilation, we cannot conclude that Section 9.4.5 is complete. The applicant is required to provide the ambient conditions required for operation of the hydrogen recombiner equipment located outside of containment. Additionally, the applicant should indicate the safety-related system that will maintain these conditions following a LOCA and loss of offsite power.]

We will report on the resolution of this matter in a supplement to this SER.

STEAM AND POWER CONVERSION SYSTEM

10.3.1 Main Steam Supply System (Up to and Including the Main Steam Block Valve)

The steam generated in the reactor vessel is routed to the high-pressure turbine by means of four main steam lines. Each main steam line contains two main steam isolation valves (MSIVs) and a shut-off (block) valve, thus assuring main steam line isolation in the event of a steam-line break outside containment and a concurrent single failure of an MSIV. One MSIV is located immediately inside of the drywell and the other immediately outside containment. The shutoff valve is located downstream of the outboard MSIV immediately before the steam lines leave the auxiliary building. The main steam isolation valves are designed to provide positive isolation against steam flow associated with a main steam-line break. They are pneumatic or spring-operated (to close) fast closing (3 to 10 seconds) valves. Operating air is supplied to the valves from the instrument air system, and a seismic-Category-I air accumulator provides back-up operating air for each valve in the event of loss of the normal instrument air supply. The MSIVs are designed to withstand the dynamic forces under the postulated steamline break flow conditions. The main steam shut-off valves are leak-tight motor-operated gate valves and are powered from separate emergency buses. They are manually actuated from the control room.

The main steam supply lines including the MSIVs and shut-off valves are seismic Category I from the reactor to the auxiliary building wall and are designed to Quality-Group-A criteria from the reactor through the outermost MSIV and Quality-Group-B criteria from the outermost MSIV through the shut-off valve to the auxiliary building wall. The lines pass through the drywell, containment and auxiliary buildings which are seismic-Category-I, flood- and tornado-protected structures. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," are satisfied for these portions of the main steam supply system.

The MSIVs which are required to function in order to assure main steam isolation are protected against the effects of high-energy pipe breaks and are qualified to function in the expected steam environment resulting from a main steam-line break. Refer to Sections 3.6.1 and 3.11 of this SER for further discussion on environmental qualification of essential equipment. This equipment is located in tornado-missile-protected structures (the steam tunnel) and is separated from the effects of internally generated missiles. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," are satisfied. For compliance with the guidelines of Regulatory Guide 1.115, "Protection Against Low Trajectory Tubine Missiles" refer to Section 3.5.1.3 of this SER.

Based on the above, we conclude that the main steam supply system from the reactor to the turbine building wall meets the requirements of General Design Criteria 2 and 4 with respect to protection against seismic events, floods, tornadoes, missiles and environmental effects, and the guidelines of Regulatory Guides 1.29 and 1.117, relating to the system's seismic classification, protection against tornado missiles and high- and moderate-energy pipe breaks and is, therefore, acceptable.

10.4.2 MAIN CONDENSER EVACUATION SYSTEM

10.4.2.1 System Description

The main condenser evacuation system is designed to (1) establish a vacuum on the condenser during startup, (2) remove noncondensible gases from the main condenser and discharge them to the gaseous radwaste system, and (3) condense any steam removed from the condenser with noncondensible gases and return the condensate to the condenser.

The major components are the mechanical vacuum pumps and the steam jet air ejectors. The main condenser evacuation system is designed to minimize the potential for explosion in the piping upstream of the catalytic recombiners in the offgas system by maintaining sufficient dilution steam in the steam jet air ejector discharge to limit the hydrogen concentration to less than four percent by volume. The steam jet air ejectors, intercoolers, and the offgas system (Section 11 of this report) are designed to withstand an explosion in the offgas system. The hydrogen concentration at the outlet of the second stage air ejector will be maintained below four percent hydrogen in air by the addition of dilution steam. On indication of low steam pressure or low steam flow, the operating steam jet air ejector will be removed from service and the standby air ejector activated. A hydrogen analyzer will be provided on the outlet of each recombiner to preclude the buildup of explosive mixtures.

10.4.2.2 Evaluation Findings

The Effluent Treatment Systems branch has verified that sufficient information has been provided and that the main condenser evacuation system design is adequate to support the following:

The main condenser evacuation system includes equipment and instruments to establish and maintain condenser vacuum and to prevent an uncontrolled release of radioactive materials to the environment. The scope of our review included the system capability to transfer radioactive gases to the gaseous waste processing system or ventilation exhaust systems, the design provisions incorporated to monitor and control releases of radioactive materials in gaseous effluents in accordance with General Design Criteria 60 and 64 and the quality group classification of equipment and components used to collect gaseous radioactive wastes relative to the guidelines of Regulatory Guide 1.26. We have reviewed the applicant's system descriptions, piping and instrumentation diagrams, and design criteria for the components of the main condenser evacuation system. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the main condenser evacuation system to the applicable regulations, regulatory guides, and industry standards referenced above. Based on our evaluation, we find the proposed main condenser evacuation system acceptable.

/5-4

10.4.3 TURBINE GLAND SEALING SYSTEM

10.4.3.1 System Description

The turbine gland sealing system is designed to provide a continuous supply of "clean" steam to main turbine shaft seals, the stem packings of stop valves, control valves, combined intermediate valves and bypass valves, the shaft seals of the reactor feed pump turbines and the stem packing of the reactor feed pump turbines, and of stop and control valves. This sealing steam is used to prevent air leaking into the steam cycle and radioactive steam leaking out of the steam cycle into the Turbine Building.

10.4.3.2 Evaluation Findings

Our review included the source of sealing system steam and the provisions incorporated to monitor and control releases of radioactive material in gaseous effluents in accordance with General Design Criteria 60 and 64, and Regulatory Guide 1.26.

The basis for acceptance in our review has been conformance of the applicants' design, design criteria, and design bases for the turbine gland sealing system to the applicable regulations referenced above. Based on our evaluation, we find the design of the steam seal system to be acceptable.

10.4.5 Circulating Water System

The nonsafety-related (non-seismic Category I, Quality Group D) circulating water system (CWS) is designed to remove the heat rejected from the main condenser to the atmosphere via one natural draft cooling tower per unit. The CWS is not required to maintain the reactor in a safe shutdown condition or mitigate the consequences of accidents.

The applicant provided the results of an analysis of the effects of possible flooding as a result of a postulated failure of a circulating water system expansion joint or butterfly valve. Flooding in the affected unit turbine building and adjacent condensate demineralizer building will result from either of the above postulated failures. The analysis further postulated that the entire volume of the system including the cooling tower basin would be emptied into the buildings. The resulting water level, 591.1 ft, would be below any doors or unsealed penetrations into the adjacent auxiliary building, thereby verifying that a total failure in the circulating water system will not result in flooding which would compromise plant safety.

Since no safety-related equipment is affected by a postulated failure in the CWS, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," with respect to protection of safety-related systems from failure of nonsafety-related systems are satisfied.

Indication of leakage or potential failure in CWS components is provided to operators in the control room. First indication would be an alarm in the control room generated by a level switch mounted just above the turbine building basement floor. If the water level should continue to rise to a second "verification" level switch located 3 ft above the floor, a second alarm would be sounded in the control room prompting operator shutdown and isolation of the system. A set of three level switches located 5 ft above the floor would initiate, on a two-out-of-three logic, automatic circulating water pump trip, pump discharge valve closure, and closure of all the condenser water box valves. CWS performance is monitored by pressure and temperature indicators in the control room.

Based on our review, we conclude that the circulating water system meets the requirements of General Design Criterion 4 with respect to protection of safety-related systems from failures in nonsafety-related systems. We therefore conclude that the circulating water system is acceptable.

~~On the basis of the above evaluation, we find that the proposed process sampling system meets the relevant requirements of 10 CFR Part 20, & 20.1(c), General Design Criteria 1, 2, 12, 14, 26, 60, 63, and 64 in Appendix A to 10 CFR Part 59, and Regulatory Guides 8.8, 1.26 and 1.29 and, therefore, is acceptable, pending confirmatory review of the amended FSAR.~~

*10.4.6 Condensate Filter Demineralizer System

I. INTRODUCTION

The condensate filter demineralizer system removes corrosion products, condenser inleakage impurities, and impurities present in the condensed steam. The system consits of the necessary piping, valves, appurtenances, and instrumentation to control the condensate impurity concentration during plant operation. Eight filters, in series with six deep bed demineralizers are provided to polish the condensate flow; six of the filters and five

* This evaluation is based upon applicant information through FSAR amendment 3 and by telecon and telex and is subject to confirmatory review of commited FSAR amendments.

demineralizers are normally in operation to process 100% condensate flow with the remaining units in standby.

II. EVALUATION AND FINDINGS

The system description and piping and instrumentation diagram were reviewed in accordance with SRP Section 10.4.6. Each demineralizer has an effluent resin strainer to prevent resin carryover with the condensate. The limits for the conductivity, chloride concentration, silicon, pH and 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. Sample line valves are provided in each demineralizer effluent line and the influent and effluent headers to permit analysis of the water quality.

To ensure that water quality is maintained within the limits of Regulatory Guide 1.56, Revision 1, Table 2, each deep bed demineralizer has an effluent conductivity cell and a conductivity cell which is located at two-thirds of bed depth. When the latter conductivity cell alarms (0.15 umho/cm), indicating that the resin bed is two-thirds exhausted, the unit is removed from service and regenerated. The total capacity of each new batch of ion exchange resin will be analytically verified by the resin supplier and rechecked annually with approved ASTM procedures.

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

10.4.7 Condensate and Feedwater System

The condensate and feedwater system includes all components and equipment from the condenser outlet to the connection to the reactor vessel and to the heater drain system. The system serves no safety function and is therefore classified as non-safety-related (Quality Group D, non-seismic Category I). However, the portion of the system between the reactor vessel and the auxiliary building wall (in the steam tunnel) is safety-related and designed to seismic-Category-I, Quality-Group-A criteria from the reactor to the outboard containment isolation valve, and seismic-Category-I, Quality-Group-B criteria from the outboard containment isolation valve through the feedwater shut-off valve to the auxiliary building wall in order to assure feedwater system isolation under accident conditions. Each main feedwater line contains a spring-closing check valve held open by air pressure during normal operation as the outboard containment isolation valve, an inboard isolation check valve, and a motor-operated shut-off valve powered from a separate emergency bus. Thus, feedwater isolation is assured in the event of a single failure in any isolation valve. The safety-related portion of the system is located in the seismic-Category-I, flood- and tornado-protected auxiliary building. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," are satisfied. The auxiliary building also provides protection against tornado missiles. The essential equipment is separated from the effects of internally generated missiles and is qualified to function in a steam-line break environment. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," are satisfied. Refer to Sections 3.6.1 and 3.11 of this SER for further discussion of environmental qualification of essential equipment and protection against postulated piping failures. The feedwater system is not shared between units; therefore, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable.

The feedwater system is not required to transfer heat under accident conditions and, therefore, General Design Criteria 44, "Cooling Water," 45, "Inspection of Cooling Water Systems," and 46, "Testing of Cooling Water Systems," are not applicable.

Based on the above, we conclude that the safety-related portion of the condensate and feedwater system meets the requirements of General Design Cirteria 2 and 4 with respect to its protection against natural phenomena, missiles and environmental effects, and meets the guidelines of Regulatory Guide 1.29 with respect to its seismic classification and is, therefore, acceptable.

SECTION 11

RADIOACTIVE WASTE MANAGEMENT

11.0 RADIOACTIVE WASTE MANAGEMENT

The review performed under this section pertains to the applicant's design provisions for controlling, processing, packaging, or minimizing radioactive plant effluents resulting from normal operation, and anticipated operational occurrences. Projected annual releases of liquid and gaseous radioactive effluents and projected volumes of solidified processed radioactive wastes calculated by the applicant are compared with data from operating plants having similar nuclear steam supply systems and comparable waste treatment systems. The applicant's provisions for the instrumented monitoring and for the sampling and analysis of liquid and gaseous radioactive effluents are compared to guidelines of appropriate Regulatory Guides.

The radioactive waste management systems provided by the applicant are essentially as described in Section 11.0 of the Safety Evaluation Report for the construction permit stage (SER-CP) dated July 1974. The liquid radioactive waste systems process wastes from equipment and floor drains, phase separator decantation, demineralizer backwash, demineralizer regenerants, decontamination and laboratory wastes, and laundry and shower wastes. The gaseous radioactive waste systems include a refrigerated charcoal delay system to allow decay of short-lived noble gases removed from the main condenser and treatment of building ventilation exhausts through high efficiency particulate air filters and charcoal adsorbers to reduce releases of radioactive materials to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 20 and 10 CFR Part 50.34a. The solid radioactive waste system provides for the volume reduction, solidification, packaging, and storage of radioactive wastes generated during station operation prior to shipment offsite to a licensed facility for disposal.

The process and effluent radiological monitoring and sampling systems provided by the applicant for normal operation and anticipated operational occurrences are essentially as described in Section 11.5 of the Safety Evaluation Report for the construction permit stage (SER-CP), dated July 1974. These systems have been augmented by the addition of high range effluent sampling and monitoring systems to accommodate the maximum calculated releases which could occur as the result of an accident, in accordance with the guidance of NUREG-0737 and Regulatory Guide 1.97.[1]

---

[1] Evaluation pending submittal of design data; licensee has committed to meeting provisions of Regulatory Guide 1.97 and NUREG-0737.

11.1 LIQUID RADIOACTIVE WASTE TREATMENT SYSTEMS

The review performed under Standard Review Plan 11.2 pertains to the following system design factors.

- System design
- System design objectives
- Design criteria
- Methods of treatment
- Expected releases
- Parameters used in calculating releases
- P&IDs and flow diagrams
- Equipment system and component design capacities
- Expected system flows and radioactivity concentrations
- Expected component decontamination factors
- System holdup time (as applicable)
- Availability of standby equipment
- Alternate processing routes
- System interconnections
- Quality group classification
- Special design provisions

We review the liquid radwaste processing systems and components with respect to the above factors on an individual component basis, on the basis of interaction between components comprising a system, and on interaction between systems. Our review must arrive at the conclusion that there is adequate assurance that a given system will perform its design function under all postulated combinations of normal operating conditions while keeping discharges to the environment "as low as is reasonable achievable". Our review did not identify any open items.

We reviewed the applicant's source term for liquid radioactive effluents, which was calculated using the methods described in NUREG-0016 (1976). Based on the applicant's list of parameters employed in the calculation, on comparison of the applicant's source term with source terms previously calculated by the staff for similar plants employing similar liquid radwaste treatment system components and design features, and on comparison with reported annual average releases from similar plants, we conclude that the applicant's source term is representative of liquid radioactive effluents which can be expected to be released from the Perry Nuclear Power Plant, Unit No. 1; therefore, the staff has adopted the applicant's source term for its use in determining the environmental impact of liquid releases.

The applicant has employed components and system designs for his liquid radwaste treatment systems which are consistent with components and systems used in operating plants and which have demonstrated their efficiency ratings, capacity ratings, and availability factors in extensive operational use. The capacities of system components are consistent with the size of the plant and with the expected volumes of waste to be processed. Processing characteristics and radioactive decontamination efficiencies of the systems and components used have been demonstrated or verified in operational use.

Our findings are as follows:

The liquid radwaste treatment systems include the equipment and instrumentation to control the release of radioactive materials in liquid effluents.

In our evaluation, we considered releases of radioactive materials in liquid effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant for each reactor on the Perry site. We determined that the proposed liquid radwaste treatment systems will be capable of maintaining releases of radioactive materials in liquid effluents such that the calculated individual doses in an unrestricted area from all pathways of exposure are less than 3 millirems to the total body and 10 millirems to any organ.(2)

We considered the capabilities of the proposed liquid radwaste treatment system to meet the anticipated demands of the plant due to anticipated operational occurrences and have concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant.

We reviewed the applicant's quality assurance provisions for the liquid radwaste systems, the quality group classifications used for system components, and the seismic design applied to structures housing these systems. The design of the systems and structures housing these systems meet the criteria as set forth in Regulatory Guide 1.143.

(2)Pending determination by RAB. Radiological Assessment Branch.

We reviewed the provisions incorporated in the applicant's design to control the releases of radioactive materials in liquids due to inadvertent tank overflows and concluded that the measures proposed by the applicant are consistent with the criteria as set forth in Regulatory Guide 1.143.

Based on the foregoing evaluation, we conclude that the proposed liquid radwaste treatment system is acceptable. The basis for acceptance has been conformance of the applicant's design, design criteria, and design bases for the liquid radioactive waste treatment systems to the Commission's regulations and to applicable Regulatory Guides, as referenced above.

11.2 GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEMS

The review performed under Standard Review Plan 11.3 pertains to the following system design factors:

- System design
- System design objectives
- Design criteria
- Methods of treatment
- Expected releases
- Parameters used in calculating releases
- P&IDs and flow diagrams
- Equipment system and component design capacities
- Expected system flows and radioactivity concentrations
- Expected component decontamination factors
- System holdup time (as applicable)
- Availability of standby equipment
- Alternate processing routes
- System interconnections
- Quality group classification
- Special design provisions

We reviewed the gaseous radwaste and building ventilation exhaust treatment systems and components with respect to the above factors on an individual component basis, on the basis of interaction between components comprising a system, and on interaction between systems. Our review must arrive at the conclusion that there is adequate assurance that a given system will perform its design function under all postulated combinations of normal operating conditions, while keeping discharges to the environment "as low as is reasonably achievable". Our review identified one open item.

We reviewed the applicant's source term for gaseous radioactive effluents which was calculated using the methods described in NUREG-0016 (1976). Based on the applicant's list of parameters employed in the calculation, on comparison of the applicant's source term with source terms previously calculated by the staff for similar plants employing similar gaseous radwaste and building ventilation exhaust treatment system components and design features, and on comparison with reported annual average releases from similar plants, we concluded that the applicant's source term is representative of gaseous radioactive effluents which can be expected to be released from the Perry Nuclear Power Plant, Unit No. 1; therefore, the staff has adopted the applicant's source term for its use in determining the environmental impact of gaseous releases.

The applicant has employed components and system designs for his gaseous radwaste and building ventilation exhaust treatment systems which are consistent with components and systems used in operating plants and which have demonstrated their efficiency ratings, capacity ratings, and availability factors in extensive operational use.

The applicant states that the design of the refrigerated charcoal offgas system meets the requirements of Regulatory Guide 1.143. For combustible gas control, redundant hydrogen analyzers have been provided upstream of the delay beds and downstream of the recombiners. Process steam is used for dilution at all times and is sized to keep gases from the air ejector below the flammable limit. Pressure vessels in the system are designed to 350 psig static pressure and piping and valving are designed to resist dynamic pressures encountered in long runs of piping at design temperature. The system is designed to operate at less than 2 psig (17 psia) during normal operation and at a maximum of 7 psig (22 psia) during startup. During normal operation, the design meets the SRP 11.3 criteria of 20 times the operating absolute pressure for systems designed to withstand the effects of a hydrogen explosion.

In addition to the ESF atmosphere cleanup systems described under Section 6.5.1 of this draft SER, the applicant has provided filtered ventilation exhaust treatment systems for the following facilities:

- Controlled Access and Miscellaneous Equipment Areas HVAC System
- Offgas Building Exhaust System
- Radwaste Building Exhaust System
- Auxiliary Building Ventilation System
- Reactor Building Containment Vessel and Drywell Purge

For each system, treatment provisions include, sequentially, roughing or prefilters, upstream HEPA filter, charcoal adsorber, and downstream HEPA filter. The applicant has not, however, specified his design criteria for these systems and has not provided information comparing his design to the guidelines of Regulatory Guide 1.140. In Table 3.2.1 of the FSAR, the applicant shows the design classification to be seismic Category I and construction codes in accordance with ERDA 76-21, ANSI 509, RDT M16-IT, and ANSI N101.1. The FSAR does not contain

specific information to determine conformance to R.G. 1.140; however, if the design conforms to N509, the applicant should have no deviations
460.15 from R.G. 1.140. <u>OPEN ITEM</u>: Applicant should provide information relative to conformance to R.G. 1.140 guidelines.

The capacities of system components are consistent with the size of the plant and with the expected volumes of waste to be processed. Processing characteristics, radioactive decontamination efficiencies, and holdup or decay times of the systems and components have been demonstrated or verified in operational use.

Subject to resolution of the one open item identified above, our findings are expected to be as follows:
The gaseous radwaste treatment system includes the equipment and instruments to control the release of radioactive materials in gaseous effluents.

In our evaluation, we considered releases of radioactive material (noble gases, radioiodine and particulates) in gaseous effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant for each reactor on the Perry-1 site. We have determined that the proposed gaseous radwaste treatment systems are capable of maintaining releases of radioactive materials in gaseous effluents such that the calculated individual doses in an unrestricted area from all pathways of exposure are less than 5 mrem to the total body and 15 mrem to any organ and that releases of radioiodine and radioactive material in particulate form are less than 15 mrem to any organ.(4)

We also considered the potential consequences resulting from reactor operation with a fission product release rate consistent with a noble gas release rate to the reactor coolant of 100 uCi/MWt-sec at 30 minutes decay determined that under these conditions, the concentrations of radioactive materials in gaseous effluents in unrestricted areas will be a small fraction of the limits in 10 CFR Part 20.(5)

We considered the capabilities of the proposed gaseous radwaste treatment systems to meet the anticipated demands of the plant due to anticipated operational occurrences and have concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant.

We reviewed the applicant's quality assurance provisions for the gaseous radwaste systems, the quality group classifications used for system components, the seismic design applied to the design of the system, and of structures housing the radwaste systems. The design of the system and structures housing these systems meet the criteria as set forth in Regulatory Guides 1.143 and 1.140.

(4) Pending determination by RAB.

(5) Pending determination by RAB.

We reviewed the provisions incorporated in the applicant's design to control releases due to hydrogen explosions in the gaseous radwaste system and conclude that the measures proposed by the applicant are adequate to prevent the occurrence of an explosion or to withstand the effects of an explosion.

Based on the foregoing evaluation, we conclude that the proposed gaseous radwaste treatment system is acceptable. The basis for acceptance has been conformance of the applicant's designs, design criteria, and design bases for the gaseous radwaste treatment system to the applicable regulations and Regulatory Guides referenced above.

11.3 SOLID RADIOACTIVE WASTE MANAGEMENT SYSTEMS

The review performed under Standard Review Plan 11.4 pertains to the following system design factors:

- System design
- System design objectives
  (1) Expected and design volumes of waste
  (2) Activity and expected radionuclide distribution
- Equipment design capacities
- Parameters employed in design
- Piping and instrumentation diagrams and flow diagrams
- Expected chemical content, flows and radionuclide concentrations
- Expected volumes requiring re-processing or further treatment
- Methods for solidification, solidifying agent employed
- Process control program
- Type and size of solid waste containers
- Method of filling and handling containers
- Monitoring and decontamination
- Packaging and storage
- Quality group classification
- Special design provisions

We reviewed the solid radioactive waste management systems and components with respect to the above factors on an individual component basis, on the basis of interaction between components comprising a system, and on interaction between systems. Our review must arrive at the conclusion that there is adequate assurance that a given system will perform its design function under all postulated combinations of normal operating conditions. Our review did not identify any open items.

We reviewed the applicant's estimates of solid radioactive waste volumes and the attendant estimates of radioactivity content. We compared the applicant's estimates of waste volume and content with the staff's expected volumes and content for similar plants, which are based on average annual volumes and content reported by operating plants. We found no significant difference between the applicant's estimates and the staff's expected volumes and content; therefore, we have accepted the applicant's estimates for use in our evaluation of the solid radwaste systems.

The applicant has employed components, system designs, and design criteria for his solid radwaste systems which are consistent with components and systems used in operating plants and which have demonstrated their efficiency ratings, capacity ratings, and availability factors in extensive operational use. The applicant has selected a cement-silica solidification system for conversion of wet wastes to a solid material. A compacting system will be used for volume reduction of dry solid wastes where practicable. 55-gallon steel drums will be employed for both solidification of wet wastes and packaging of dry wastes; however, larger containers can also be used and are compatible with the packaging systems, except that the large containers would not be used for compacted dry wastes.

460.16 It is noted that the applicant has not described a process control program for the purpose of providing assurance that solidification of wet wastes will meet the applicable requirements for packaging, handling, shipping, and disposal, nor has he committed to such a program. While this is not a subject to be addressed in the SER, such a program will be a requirement of the Technical Specifications. (Reference Branch Technical Position BTP-ETSB 11-3, attached to SRP 11.4.)

Applicant has provided a shielded storage vault of approximately 16,200 ft^3 (usable space) for the on-site storage of solidified waste and a shielded volume of approximately 5,000 ft^3 (usable space) for storage of compacted dry waste. The applicant projects an average monthly generation rate of 8,100 ft^3 per month of solidified wastes and 1,000 ft^3 per month of compacted wastes. Assuming a 60% space utilization factor, the storage volumes would accommodate approximately a one-month generation of solidified waste and a 3-month generation of compacted waste. Since Branch Technical Position BTP-ETSB 11-3 provides for accommodating at least 30-days generation of wastes, the applicant's provisions are acceptable; however, in view of recent history in the availability of disposal sites, it is suggested that it would be prudent for the applicant to make contingency plans for additional on-site storage volume. (Note: This latter statement is merely a suggestion, not an identified open item of the draft SER.)

The applicant has committed to following appropriate federal and state regulations relative to the packaging and shipment of solid wastes to an approved offsite burial facility.

Our findings are as follows:

"The solid waste system (SWS) includes the equipment and instrumentation used for the solidification, packaging, and storage of radioactive wastes prior to shipment offsite for burial. The scope of the review of the SWS includes line diagrams of the system, piping and instrumentation diagrams (P&IDs), and descriptive information for the SWS and for those auxiliary supporting systems that are essential to the operation of the SWS. The applicant's proposed design criteria and design bases for the SWS, and the applicant's analysis of those criteria and bases have been reviewed. The capability of the proposed system to process the types and volumes of wastes expected during normal operation and anticipated operational occurrences in accordance with General Design Criterion 60, provisions for the handling of wastes relative to the requirements of 10 CFR Parts 20 and 71 and of applicable DOT regulations, and the applicant's quality group classification and seismic design relative to Regulatory Guide 1.143, have also been reviewed. The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the solid radwaste system to the regulations and the guides referenced above, as well as to staff technical positions and industry standards. Based on the foregoing evaluation, we conclude that the proposed solid radwaste system is acceptable."

11.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING INSTRUMENTATION AND SAMPLING SYSTEMS

The review performed under Standard Review Plan 11.5 pertains to the systems provided by the applicant for the purposes of detecting and measuring concentrations of radioactive materials within plant process systems and in plant gaseous and liquid effluents. Our review was concerned with the applicant's provisions for the instrumented monitoring of specific plant process streams and plant effluent release paths and for the sampling and laboratory analysis of materials in specific plant effluent release paths. Our review included: the type, location, and range of instrumented monitoring; actuation of control or isolation devices by instrumented monitoring; location of sensors relative to process system piping or ducts and relative to effluent piping, ducts, diffusers, stacks, or other release points; special provisions for sampling or monitoring, e.g., isokinetic sampling; provisions for laboratory analysis of collected samples; location of sampling points; provisions for monitoring releases from postulated accidents; descriptions or procedures for calibration, maintenance and inspection; and layout drawings, P&IDs, and process flow diagrams showing sensor or instrument locations.

We reviewed the process and effluent radiological monitoring instrumentation and sampling systems with respect to the factors listed above. Our review must arrive at the conclusion that there is adequate assurance that a given monitor or -- in other cases -- provisions for sampling and laboratory analysis, will be capable of providing the intended design function or purpose under all postulated conditions of normal operation, including anticipated operational occurrences, and that either adequate range capacity or alternative high-range capability is provided for the detection and measurement of radioactivity levels in effluents during and following accidents.

460.17 With respect to the monitoring, sampling and analysis of process and effluent streams under accident conditions, the applicant has not as yet made his detailed submittal on items identified in Sections II.F.1-1 and II.F.1-2 of NUREG-0737; therefore, these must be identified as open items.

No open items were identified with respect to monitoring, sampling, and analysis of process or effluent systems for normal operations, including anticipated operational occurrences.

The applicant has employed sensors, sample collecting and conditioning systems, and sample analysis capability for his process and effluent radiological monitoring and sampling systems which are consistent with those employed in operating plants and which have demonstrated their effectiveness in extensive operational use. The applicant has provided instrumented monitoring for effluent gaseous and liquid discharge paths in accordance with the guidelines of Regulatory Guide 1.21. Similarly, the applicant has provided instrumented monitoring at many points in the in-plant liquid and gaseous radioactive process systems for the purposes of quantifying and controlling radioactivity concentrations in plant systems in accordance with the guidance in Standard Review Plan 11.5.

Subject to resolution of the open items noted above for NUREG-0737 items, our findings are expected to be as follows:

"The process and effluent radiological monitoring and sampling systems include the instrumentation for monitoring and sampling radioactively contaminated liquid, gaseous, and solid waste process and effluent streams. Our review included the provisions proposed to sample and monitor all station effluents in accordance with General Design Criterion 64, the provisions proposed to provide automatic termination of effluent releases and assure control over discharges in accordance with General Design Criterion 60, the provisions proposed for sampling and monitoring plant waste process streams for process control in accordance with General Design Criterion 63, the provisions for conducting sampling and analytical programs in accordance with the guidelines in Regulatory Guide 1.21, and Regulatory Guide 4.15, and the provisions for sampling and monitoring process and effluent streams during postulated accidents in accordance with the guidelines in Regulatory Guide 1.97. The review included piping and instrument diagrams and process flow diagrams for the liquid, gaseous, and solid radwaste systems, and for ventilation systems, and the location of monitoring points relative to effluent release points as shown on the site plot diagrams."

"The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the process and effluent radiological monitoring, instrumentation and sampling systems to the applicable regulations and guides, as indicated above, as well as to staff technical positions and industry standards. Based on our evaluation, we find the proposed systems to be acceptable."

RADIATION PROTECTION

SECTION 12

PERRY NUCLEAR POWER PLANT
''NITS 1 AND 2
SAFETY EVALUATION REPORT

We have evaluated the proposed radiation protection program presented in Chapter 12 of the Perry Nuclear Power Plant (PNPP) FSAR. This FSAR was submitted by the Cleveland Electric Illuminating Company (CEI). The radiation protection measures at Perry are intended to ensure that internal and external radiation exposure to plant personnel, contractors, and the general population due to plant conditions, including anticipated operational occurrences, will be within applicable limits of 10 CFR Part 20, and will be as low as is reasonably achievable (ALARA).

The basis of our acceptance of Perry's Radiation Protection Program is that doses to personnel will be maintained within the limits of 10 CFR 20, "Standard for Protection Against Radiation." The applicant's radiation protection designs and program features are consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable" (Rev. 3). Some of the radiation protection measures which the applicant will use at Perry include: Ventilation systems designed for easy access and service to minimize doses during maintenance, decontamination and filter change; use of remote handling equipment; location of radiation components in separately shielded cubicles; and training of personnel in radiation protection. The applicant's use of these and other radiation protection features will help to ensure that occupational radiation exposures are maintained as low as reasonably achievable, both during plant operation and during decommissioning.

On the basis of our review of the Perry Nuclear Power Plants (PNPP) FSAR, we have concluded that the radiation protection measures incorporated in the design will provide a reasonable assurance that occupation doses will be maintained as low as is reasonably achievable and below the limits of 10 CFR

Part 20. These radiation design features are consistent with the guidelines of Regulatory Guide 8.J (Rev. 3).

12.1 Ensuring that Occupational Radiation Exposures are as low as is Reasonably Achievable (ALARA)

The applicant provides a management commitment to assure that the Perry Nuclear Power Plant (PNPP) will be designed, constructed, and operated in a manner consistent with Regulatory Guides 8.8 "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as low as is Reasonably Achievable"; 8.10 "Operating Philosophy for Maintaining Occupational Radiation Exposures as low as is Reasonably Achievable"; and 1.8 "Personnel Selection and Training" (Rev. 1). The ALARA philosophy was applied during the initial design of the plant, since then the applicant has continued to review, update, and modify the plant design and construction phases. The plant's ALARA committee periodically reviews, updates, and modifies all plant design features, and maintenance features as appropriate, using exposure data and experience gained from operating nuclear power plants. This is done to assure that occupational exposures will be kept as low as is reasonably achievable in accordance with Regulatory Guide 8.8 criteria. Therefore, the policy considerations are acceptable.

The objective of the plant's radiation protection design is to maintain individual doses and total person-rem doses to plant workers, including construction workers, and to members of the general public as low as is reasonably achievable, and to maintain individual doses within the limits of 10 CFR Part 20. Within restricted areas all plant sources of direct radiation and airborne radioactive contamination are considered in our review.

To reduce radiation exposures the applicant has utilized feedback information obtained from plants currently operational. The following type of feedback information was considered useful in the plant design:

(1) Operations radiation levels.

(2) Trends in radiation levels associated with years of operation, plant type, plant size, power levels and plant design.

(3) Radiation zones as determined by occupancy requirements and actual radiation levels.

(4) Location of components with respect to plant operability.

(5) Reliability of components.

(6) Adequacy of plant layout in terms of traffic patterns, and space allocation such as around radioactive components requiring maintenance and inspections, and pipe routing.

(7) Number of plant employees associated with different tasks and the resulting person-rem doses.

Utilizing the feedback information from operating plants and following the Guidelines of Regulatory Guide 8.8, the Cleveland Electric Illuminating Company (CEI) has incorporated the following facility and equipment design considerations at Perry Nuclear Power Plant in order to satisfy the plants radiation protection design objectives:

(1) Access labyrinths are provided for rooms housing equipment that contains high radiation sources to preclude a direct radiation path from the equipment to accessible areas.

(2) Piping penetrations, ducts and voids in radiation shield walls are located to preclude the possibility of streaming from a high to low radiation area.

(3) Radioactive piping is routed through high radiation areas where practicable, or in shielding pipe cases in low radiation areas.

(4) Sufficient work area and clearance space is provided around equipment to permit ease of servicing.

(5) Natural traps which could be potential pockets of corrosion product activity are minimized in pipes and ducts by avoiding sharp bends, rough finishes and cracks.

These design considerations conform with the guidelines of Regulatory Guide 8.8 and are acceptable.

The Perry Nuclear Power Plant operational considerations included the development of a radiological training program, a radiation zoning and access control system and general guidelines for workers performing maintenance in high radiation areas. These operational considerations ensure that operating and maintenance personnel follow specific plans and procedures in order to assure that "as low as is reasonably achievable" goals are achieved in the operation of the plant. High radiation exposure operations are carefully preplanned and carried out by personnel well trained in radiation protection and using proper equipment. During such maintenance activities, personnel are monitored for exposure to radiation and contamination. Upon completion of major maintenance jobs, personnel radiation exposures are evaluated and compared with predicted person-rem exposures. The results are used to make changes in future job procedures and techniques. The plant's ALARA Committee periodically reviews radiation exposure trends to determine major changes in problem areas, and which worker groups are accumulating the highest exposures. The staff uses these reports to recommend design modifications or changes in plant procedures. The operational considerations conform to Regulatory Guides 8.8 and 8.10 and are acceptable.

We conclude that the policy considerations, design considerations and operational considerations at Perry Nuclear Power Plant are adequate to assure that occupational radiation exposures will be ALARA in accordance with Regulatory Guides 8.8 and 8.10 and are acceptable.

12.2 Radiation Sources

Section 12.2 of the Perry Nuclear Power Plant FSAR describes the sources of contained and airborne radioactivity used as inputs for the dose assessment and for the design of the shielding and ventilation systems. The methods and bases used by the applicant to estimate the source terms are also described. Additional information on source terms are described in Chapter 11.

Inside the containment during power operation, the greatest potential for personnel dose during operation is due to nitrogen-16, noble gases, and neutrons. Outside the containment and after shutdown inside the containment the primary sources of personnel exposure are fission products from fuel clad defects, and activation products, including activiated corrosion products. Almost all of the airborne radioactivity within the plant is due to equipment leakage. The fission product source terms are based on an offgas rate of 100,000 mci/sec after 30 minutes delay. The coolant and corrosion activation product source terms are based on operating experience and reactors of similar design; allowances are included for the buildup of activated corrosion products. Neutron and prompt gamma source terms are based on reactor core physics calculations and operating experience from reactors of similar design. The source terms presented are comparable to estimates by other applicants with similar design and are acceptable.

The applicant has provided a tabulation of the maximum expected radioactive airborne concentrations in equipment cubicles, corridors, and operating areas, due to equipment leakage. The bases for these leakage calculations are in accordance with Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors," and are acceptable.

The ventilation system will be designed to provide sufficient volume changes per hour in occupied areas which may contain significant airborne activity to maintain exposure to personnel ALARA. Air will be routed from areas of low potential airborne contamination to areas of increasing potential airborne contamination. The resulting estimated airborne radioactivity concentrations in frequently occupied areas will be a small fraction of 10 CFR Part 20.103 limits and are acceptable. The source terms used to develop these airborne concentration values are comparble to estimates by other applicants with similar design and are acceptable.

12.3 Radiation Protection Design Features

Section 12.3 of the Perry Nuclear Power Plant FSAR describes the features which are included in the radiation protection design of the plant to maintain exposures as low as is reasonably achievable. Separate descriptions are presented for the categories of facility design features, shielding, ventilation, and radiation and airborne monitoring instrumentation.

The applicant has provided evidence that the dose accumulating functions performed by workers have been considered in the plant design. Features have been included in the design to help maintain exposure as low as is reasonably achievable in the performance of those functions. These features will facilitate access to work areas, reduce or allow the reduction of source intensity, reduce the time required in the radiation fields, and provide for portable shielding and remote handling tools. The applicant's facility design features are consistent with the guidance of Regulatory Guide 8.8. Therefore, we conclude that the facility design features are acceptable.

The applicant has provided five radiation zones as a basis for classifying occupancy and access restrictions on various areas within the plant. On this basis, maximum design dose rates are established for each zone and used as input for shielding of the respective zones. The areas that will have to be occupied on a predictable basis during normal operations and anticipated occurrences are zoned such that exposures are below the limits of 10 CFR Part 20, and will be as low as is reasonably achievable. The zoning system and access control features also meet the posting entry requirements of 10 CFR Part 20.203 or standard NRC Technical Specifications and are consistent with Regulatory Guide 8.8.

Several features are included in the plant design and operational program to minimize the buildup of activated corrosion product, a major contribution to occupational doses. Examples would include:

(1) Hard facing materials which have high cobalt content, such as stellite, are used only where substitute material cannot satisfy performance requirements.

(2) Whenever possible packless valves are specified for systems which normally handle radioactive fluids.

(3) Piping systems are of all welded construction with minimum use of flanged and socket weld connections.

(4) Control of chemistry in the primary system.

The applicant's corrosion product control features are consistent with the guidance of Regulatory Guide 8.8 and are acceptable.

The design features incorporated by the applicant for maintaining occupational radiation doses as low as is reasonably achievable during plant operation and maintenance will also serve to maintain radiation doses as low as is reasonably achievable during decommissioning operations and are therefore acceptable.

The objectives of the plant's radiation shielding is to provide protection against radiation for operating personnel, both inside and outside the plant, and for the general public, during normal operation, including anticipated operational occurrences and during reactor accidents. The shielding was designed to meet the requirements of the radiation dose rate zone system discussed above. The following are several of the shielding design features incorporated into Perry Nuclear Power Plant:

(1) Access labyrinths are provided for rooms housing equipment that contains high radiation sources to preclude a direction radiation path from the equipment to accessible areas.

(2) Radioactive piping is routed through high radiation areas where practicable, or in shielded pipe chases in low radiation areas.

(3) Shielding is provided for all equipment which is anticipated to be normally radioactive.

(4) Piping penetrations, ducts and voids in radiation shield walls are located to preclude the possibility of streaming from a high to low radiation area.

(5) Shielding discontinuities caused by shield plugs, concrete hatch covers and shield doors to high radiation areas are provided with offsets to reduce radiation streaming.

These shielding techniques are designed to maintain personnel radiation exposure as low as is reasonably achievable. Therefore, we conclude that the shielding design objectives are acceptable.

The applicant's shielding designed methods, included the use of computer codes such as SDC, for simple source configurations such as cylinder and spheres; QAD6G, for complex geometrics, G-33B, for calculating scattering dose rates; and ANISIN for calculating the reactor neutron and gamma flux spectra through shielded walls. The applicant also used shielding information from operating nuclear plants as input data for their shield design calculations. All concrete shielding in the plant will be constructed in general compliance with Regulatory Guide 1.69, "Concrete Radiation Shields for Nuclear Power Plants." Therefore, we conclude that the shielding design methodology presented are acceptable.

Shielding for and access to the spent fuel transfer tube was reviewed. The applicant has described two post-licensing programs to be conducted by the PNPP Health Physics Unit to evaluate potential radiation streaming problems in accessible areas in the vicinity of the Inclined Fuel Transfer System (IFTS). The first program is a comprehensive shielding survey conducted with a spent fuel bundle in the IFTS tube. The second program will consist of placing TLD's in strategie locations during the period a spent fuel bundle is transferred via the IFTS. It is our position that the applicant should, in addition to these post-licensing programs, provide a plan and elevation layout drawing of the areas through which the spent fuel bundle passes and perform a radiation and shielding-design review of all accessible areas around the IFTS. This is an open item. The applicant intends to control access to maintenance areas where close access to the Inclined Fuel Transfer Tube may occur by using a key activated system. The system described in PNPP FSAR is acceptable if it is fail safe, i.e, the doors are secured even when power is lost to the system. The applicant should verify that the key activated system is fail safe.

The ventilation system at Perry Nuclear Power Plant will be designed to protect personnel and equipment from extreme thermal environmental conditions and ensure that plant personnel are not inadvertently exposed to airborne contaminants exceeding those given in 10 CFR Part 20.103. The applicant intends to maintain personnel exposures as low as is reasonably achievable by:

(1) maintaining airflow from areas of potentially low airborne contamination to areas of higher potential concentrations;

(2) ensuring negative or positive pressures to prevent exfiltration or infiltration of potential contaminants, respectively; and

(3) locating ventilation system intakes so that intake of potentially contaminated air from other building exhaust points is minimized.

These design criteria are in accordance with the guildelines of Regulatory Guide 8.8. Some examples of exposure reduction features in the ventilation system are listed below.

(1) The ventilation equipment is not located in normally high radioactive areas.

(2) Suitable access doors and service aisles are provided to permit ease in servicing and maintenance.

(3) The roughing and HEPA filters in the ESF filter trains and the HEPA filters in the non-ESF filter trains are serviced on the downstream side of the filter to minimize personnel exposure.

(4) The activated charcoal absorber is bulk loaded into the permanently installed, sealed welded and gasketless absorber section. Spent charcoal absorber material is vacuumed from the bottom of the plenums and loaded directly into drums for shipment offsite, with new charcoals absorber material being added at the top of the absorber section.

(5) Interior surfaces of ducts are designed to minimize the buildup of dust.

(6) Equipment redundancy is provided where practicable and idle equipment is isolated by dampers to control airborne contamination during maintenance operations.

We conclude that the applicant's ventilation system will keep personnel exposures at a small fraction of 10 CFR Part 20 values and therefore acceptable.

The applicant's area radiation monitoring system is designed to:

(1) monitor the radiation levels in areas where radiation levels could become significant and where personnel may be present;

(2) alarm when the radiation levels exceed preset levels to warn of excessive radiation levels; and

(3) provide a continuous record of radiation levels at key locations throughout the plant.

In order to meet these objectives, the applicant plans to use 58 area monitors located in areas where personnel may be present and where radiation levels could become significant. The area radiation monitoring system is equipped with local and remote audio and visual alarms and a facility for central recording.

The applicant has provided area radiation monitors around the fuel storage area to meet the requirements of 10 CFR 70.24 and to be consistent with the guidance of Regulatory Guide 8.12, "Criticality Accident Alarm Systems."

The design objectives of the applicant's airborne radioactivity monitoring system are:

(1) to assist in maintaining occupational radiation exposure to airborne contaminants as low as is reasonably achievable;

(2) to check on the integrity of systems containing radioactivity which are being monitored; and

(3) to warn of inadvertent release of airborne radioactivity to prevent over exposure of personnel.

The applicant will install airborne radioactivity monitors in work areas where there is a potential for airborne radioactivity. These airborne radioactivity monitors have the capability to detect 10 MPC-hours of particulate and iodine radioactivity in any compartment which has a possiblity of containing airborne radioactivity and which may be occupied by personnel. The applicant will provide portable continuous air monitors when needed to monitor air in areas not provided with fixed airborne radioactivity monitors. All airborne and area radioactivity monitors will be calibrated at regular time intervals or after major repair in accordance with plant procedures. The objectives and location criteria of Perry Nuclear Power Plants area and airborne radiation monitoring systems are in conformance with 10 CFR Part 20 and Regulatory Guides 8.2 and 8.8 and are acceptable.

The objective of the applicant's accident radiation monitoring system is to provide the capability to assess the radiation hazard in areas which may be accessed during the course of an accident. The installed instruments have emergency power supplies; and the portable instruments are placed to be readily accessible to personnel responding to an emergency. The systems are designed for use in the event of an accident in terms of 1) usable instrument range, and 2) the environment the instrument can withstand.

We conclude that the equipment and facility design features, shielding, ventilation, area radiation monitoring and airborne radioactivity monitoring systems at Perry Nuclear Power Plant are sufficient to assure that radiation exposures are as low as reasonably achievable and are acceptable.

12.4 Dose Assessment

The applicant has based the estimate of annual person-rem exposure at Perry Nuclear Power Plant on plant specific projections as to occupancy and dose rate, on experience from currently operating reactors, and improvements in design of systems to maintain in-plant radiation levels as low as is reasonably achievable. One example of design improvement at Perry is the applicant has installed a remote radioactive waste handling system.

The applicant has performed an assessment of the doses that will be received by plant and contractor personnel. This dose assessment is based upon occupancy factors, expected dose rates, expected airborne radioactivity concentrations, and estimates of the time and manpower necessary to perform the various tasks involved in plant operation. The dose assessment includes a breakdown of the annual person-rem doses associated with major functions such as normal operations, and maintenance, radwaste handling, refueling, and in-service inspection. The applicant estimated the total annual collective rem dose to plant personnel and contractors to be 404 person-rems per unit. This estimate is based on experience from operating plants and on infomation presented in NUREG-0323, "Occupational Radiation Exposure at Light Water Cooled Power Reactors, 1976," March 1978; NUREG-0322, "Ninth Annual Occupational Radiation Exposure Report, 1976," October 1977; Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Person-rem Estimates," May 1978; and other documents referenced by the applicant in Section 12.4, of the PNPP, FSAR.

Currently operating BWR's average 740 person-rem per unit annually, with particular plants experiencing an average lifetime annual dose as high as 1853 person-rems. These dose averages are based on widely varying yearly doses for BWR's. The difference between PNPP projected average person-rem of 404 and the current average dose can be explained by significant design improvement at Perry. Therefore, we find the bases for Perry Nuclear Power Plant exposure estimate acceptable, and consistent with the acceptable criteria in the Standard Review Plan.

The Cleveland Electric Illuminating Company provides a tabulation of the maximum expected radioactive airborne concentrations, as well as estimates of the inhalation and submersion dose equivalent rates to plant personnel. The dose equivalent rates are derived from the airborne radioactivity source terms given in Chapter 11. The applicant's assumptions and models upon which his internal and submersion dose estimates are based for occupational exposures are consistent with the staff's and are acceptable.

The applicant has also estimated the potential whole body exposure to construction workers during Unit 2 construction due to the operation of Unit 1. These estimates are only a few percent of the allowable whole body exposure given in 10 CFR 20.105.

We conclude that the applicant's dose assessment for contained sources and airborne radioactive material are comparable to estimates by other applicants with similar design and are acceptable.

12.5 Health Physics Program

Section 12.5 of the Perry Nuclear Power Plant's FSAR describes the applicant's health physics program. The description includes the radiation protection organization, equipment, instrumentation, and facilities, and the procedures for radiation protection.

The health physics program objectives are to provide reasonable assurance that the limits of 10 CFR Part 20 are not exceeded, to further reduce unavoidable exposures, and to ensure that individual and total person-rem occupational radiation exposures are maintained as low as is reasonably achievable.

The Supervisor Health Physics Unit is the Radiation Protection Manager (RPM) at Perry and is responsible for implementing and enforcing the Perry Plant health physics program. However, the ultimate responsiblity of the health physics program lies with the Plant Manager. The RPM is a member of the Plant Operations Review Committee, ALARA Committee, and has access to the Plant Manager for all radiation safety matters. The plant's radiation protection section General Supervising Engineer will act as the backup to the RPM in the event of his absence from the plant. The health physics organization, the qualification of the health physics personnel, the objectives of the health physics program, and the way in which it will be implemented are in accordance with the guidelines of Regulatory Guides 8.8, 8.10, 1.8 and NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources," and are acceptable.

The radiation protection features at Perry Nuclear Power Plant includes a Counting Room, Low and High Level Laboratory, Health Physics Offices, Chemistry Offices, Health Physics and Radiation Protection Service Room, Women's Lavatory and Locker Room, Men's Lavatory and Locker Room, Personnel Decontamination Room, portable instrument calibration area, an access control point, and laundry room. These facilities are sufficient to maintain occupational radiation exposure as low as is reasonably achievable and are consistent with the provisions of Regulatory Guide 8.8.

Equipment to be used for radiation protection purposes includes portable radiation survey instruments, personnel monitoring equipment, fixed and portable area and airborne radioactivity monitors, laboratory equipment, air sampliers, respiratory protective equipment, and protective clothing. The number and types of equipment to be used is adequate and provides reasonable assurance that the applicant will be able to maintain occupational exposure as low as is reasonably achievable.

All permanent and temporary plant personnel will be assigned thermoluminescent dosimeter (TLD) badges to be worn at all times. These badges will be processed monthly, or more frequently if significant exposures are suspected. All personnel working in controlled areas are also required to wear direct reading dosimeters. The readings from these dosimeters will be used to keep a running total of an individual's dose prior to TLD processing. Plant visitors will be issued dosimeters if they require entrance into radiation control areas as per Section 20.202 of 10 CFR Part 20. Whole body counts of all plant personnel will be conducted on a scheduled basis and other bioassays will be provided when deemed necessary by the Plant's Health Physics staff in accordance with Regulatory Guides 8.26 and 8.9. All radiation exposure information will be processed and recorded in accordance with 10 CFR Part 20.

Maintenance, repair, surveillance and refueling procedures and methods used by the applicant are reviewed to assure that all plant radiation protection procedures, practices, and criteria have been considered, to assure that occupational radiation exposures will be ALARA and in accordance with Regulatory Guide 8.8. Procedures are also developed to assure that exposure limits are not exceeded by plant or visitor personnel onsite; to administer and control conditions of radiation work permits; to post radiation areas; to establish radiation access control zones; to control all radioactive material entering or leaving the plant site; and to train plant and visitor personnel in radiation protection policies and procedures and meet the requirements of Regulatory Guide 1.33.

The applicant has committed to the following Regulatory Guides in his FSAR with respect to operating his radiation protection program: 1.8, 1.16, 1.21, 1.33, 1.52. 1.69, 1.97, 1.112, 8.1, 8.2, 8.3, 8.4, 8.5, 8.7, 8.8, 8.9, 8.10, 8.12, 8.13, 8.14, 8.15, 8.19, 8.20, 8.25, and 8.26.

Based on the information presented in the FSAR and the applicant's responses to our questions, we conclude that the applicant intends to implement radiation protection program that will maintain in-plant radiation exposures within the applicable limits of 10 CFR Part 20 and will maintain exposures as low as is reasonably achievable in accordance with Regulatory Guide 8.8.

CONDUCT OF OPERATIONS

SECTION 13

DRAFT SAFETY EVALUATION REPORT INPUT

PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2

OPERATING AND MAINTENANCE PROCEDURES

AND

ANTICIPATED TRANSIENTS WITHOUT SCRAM

13.5.2 OPERATING AND MAINTENANCE PROCEDURES

A. General

¶ A review has been conducted of the applicant's plan for development and implementation of operating and maintenance procedures. The review was conducted to determine the adequacy of the applicant's program for assuring that routine operating, off-normal, and emergency activities are conducted in a safe manner. The following description and evaluation are based on information contained in the applicant's FSAR, the applicant's response to NRC TMI Action Plan Items (NUREG-0660 and NUREG-0737), and supplemental information obtained during the review.

¶ In determining the acceptability of the applicant's program, the criteria of NUREG-0800, Standard Review Plan, Section 13.5.2 were used. The review consisted of an evaluation of (1) the applicant's procedure classification system for procedures that are performed by licensed operators in the control room ~~,~~ and the classification system for other operating and maintenance procedures; (2) the applicant's plan for completion of operating and maintenance procedures during the initial plant testing phase to allow for correction prior to fuel loading; (3) the applicant's program for compliance with the guidance contained in Regulatory Guide 1.33, Rev. 2, March 1978 regarding the minimum procedural requirements for safety-related operations; (4) compliance with the guidance contained in ANSI N18.7-1976/ANS 3.2; and (5) the applicant's program for compliance with Task Action Plan (NUREG-0660) Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents", for the development of Emergency Operating Procedure Guidelines. Additionally, the applicant's program for compliance with Item I.C.1 of NUREG-0737 for the development of Emergency Operating Procedures will be reviewed and reported in a supplement to this Safety Evaluation Report.

B. Operating and Maintenance Procedure Program

The applicant has committed in the FSAR to a program in which all safety-related activities are to be conducted in accordance with detailed written and approved procedures meeting the requirements of Regulatory Guide 1.33, Rev. 2, March 1978, "Quality Assurance Program Requirements (Operation)", and ANSI N18.7-1976/ANS 3.2.

The applicant uses the following categories of procedures for those operations performed by licensed operators in the control room:

Integrated Operations

System Operations

Off-Normal Conditions

Emergency Conditions

Alarm Response

Temporary Procedures

¶ Other procedures include the following areas:

Health Physics

Instrument Calibration

Radwaste

Fuel and Technical Instructions

Maintenance

Material Control

Emergency Plan

Security Plan

Startup Tests

Surveillance

Fire Protection

¶ Our review disclosed that the applicant's program for use of operating and maintenance procedures meets the relevant requirements of 10 CFR Part 34, and is consistent with the guidance provided in Regulatory Guide 1.33 and ANSI N18.7-1976/ANS 3.2, with the exception that an acceptable date for completion of procedures has not been provided. In Amendment No. 2 to the Perry FSAR dated May 22, 1981, the applicant indicated that the procedures would be written for normal and emergency conditions ~~and associated training completed~~ to the extent practical for use during the initial test program. This is not acceptable. We require procedures to be available about six months prior to fuel load, in accordance with Standard Review Plan (NUREG-0800) Section 13.5.2.

C. Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nuclear Reactor Regulation 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. 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 three 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 Item I.C.1 of NUREG-0737.

¶ In a submittal dated June 30, 1980, the BWR Owners' Group provided a draft of generic Emergency Operating Procedure 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)). Additional information, ~~was~~ requested by the staff, ~~and~~ was submitted by the Owners' Group on January 31, 1981. The staff will complete its review of this additional information ~~is still under review~~ prior to ~~prior to the staff~~ making a final conclusion on the acceptability of the guidelines for implementation on all Boiling Water Reactors.

¶ 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 six (6) selected BWR facilities. The staff is considering changes to the guidelines and their applicability to other Operating License Applicants and Operating Reactors and anticipates that the application of the guidelines, with some modifications, will be extended to the Perry Nuclear Power Plant. To date the applicant has not committed to an acceptable program for development of Emergency Operating Procedures to meet the requirements of NUREG-0737, Item I.C.1. Therefore, the staff has not determined if a Pilot Monitoring Review of Selected Emergency Operating Procedures in accordance with TMI Action Plan Item I.C.8 is required for the Perry Nuclear Power Plant. The staff has prepared draft guidelines for long-term upgrading of Emergency Operating Procedures (NUREG-0799) in accordance with Task Action Plan Item I.C.9. These guidelines, as revised during the resolution of public comments, should be used in the preparation of the Perry Emergency Operating Procedures. Our review of the applicant's submittal for Emergency Operating Procedures will be completed prior to issuance of the operating license and will be addressed in a supplement to this Safety Evaluation Report.

In accordance with NUREG-0737, Item I.C.7, NSSS vendor review of low power testing, power ascension and emergency operating procedures is necessary to further verify adequacy of the procedures. To date the applicant has not committed to a satisfactory review of the procedures by the NSSS Vendor, General Electric Corporation, prior to implementation of these procedures. The staff will confirm that this review is completed prior to issuance of an operating license.

15.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM

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 reduce 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 ATWS mitigation capability described in Volume 3 of NUREG-0460. Subsequently, we recommended to the Commission that rulemaking be used to determine any future modifications necessary to resolve ATWS concerns as well as the required schedule for implementation of such modifications. Perry Nuclear Power Plant, Units 1 and 2 are 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 ATWS. As a prudent course, to further reduce the risk from ATWS 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 ATWS 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 ATWS event.

2. The emergency operating procedure should describe actions to be taken in the event of an ATWS 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.

The GE Owners' Group is currently developing a set of Reactivity Control Guidelines, which will incorporate the above steps for mitigating ATWS events. The applicant's procedure for mitigating ATWS will be reviewed under the emergency operating procedure program as described in Section 13.5.2. The results of the staff review will be reported in a supplement to this Safety Evaluation Report.

Section 13.6.1

PERRY NUCLEAR POWER PLANT EMERGENCY PREPAREDNESS

The Cleveland Electric Illuminating Company submitted a revised Emergency Plan, dated May 22, 1981, as Appendix 13A in Amendment 2 to the Final Safety Analysis Report. This Plan, currently under review by the NRC staff, is required to meet the new regulations for emergency planning which were published in the Federal Register on August 19, 1980 and became effective on November 3, 1980.

The regulations contain a revised Appendix E to 10 CFR Part 50, "Emergency Planning and Preparedness for Production and Utilization Facilities", which establishes minimum requirements for an acceptable state of onsite emergency preparedness, and a new section 50.47, "Emergency Plans", which specifies standards which must be met for both onsite and offsite emergency response. This latter section incorporates the joint NRC/FEMA standards for use in evaluating State and local radiological emergency plans and preparedness.

NRC and the Federal Emergency Management Agency have agreed that FEMA will make a finding and determination as to the adequacy of State and local government emergency response plans. NRC will determine the adequacy of the applicant's emergency response plans with respect to the standards listed in Section 50.47(b) of 10 CFR Part 50, the requirements of Appendix E to 10 CFR Part 50, and the guidance contained in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" dated November, 1980. After the above determinations by NRC and FEMA, the staff will make a finding in the licensing process as to the overall and integrated state of preparedness. In accordance with Section 50.47(a) of 10 CFR Part 50, an operating license will not be issued unless our overall finding is such that the state of onsite and offsite emergency preparedness provides reaso e assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

ENCLOSURE

Appendix 13A

50-440/441

DRAFT SAFETY EVALUATION REPORT
PERRY NUCLEAR POWER PLANT

The applicant has submitted security plans entitled "Perry Nuclear Power Plant Security Plan," "Perry Nuclear Power Plant Safeguards Contingency Plan," and "Perry Nuclear Power Plant Security Force Training and Qualification Plan," for protection against radiological sabotage.

As a result of our evaluation, certain portions of each of the above mentioned plans were identified as requiring additional information or upgrading to satisfy the requirements of 10 CFR 73.55 and Appendices B and C of 10 CFR Part 73. The plans, when revised in accordance with our written comments, will be considered to meet these requirements. We are awaiting submittals from the licensee in this regard.

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

The identification of vital areas and measures used to control access to these areas, as described in the plan, may be subject to amendments in the future 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 plans are being withheld from public disclosure in accordance with the provisions of 10 CFR 73.21.

INITIAL TEST PROGRAM

ACCIDENT ANALYSIS

SECTION 15

SECTION 15 -OPEN AREAS

Transient and Accident Analysis (15)

9. We require the applicant to justify that operation with partial feedwater heating to extend the cycle beyond the normal end of cycle would not result in a more limiting change in MCPR than that obtained using the assumption of normal feedwater heating. Otherwise, operation with partial feedwater heating will be prohibited.

10. The applicant must consider the generator load rejection without bypass and the turbine trip without bypass as events of moderate frequency and take them into account in the determination of operating MCPR.

11. We require additional information from the applicant explaining why the generator load rejection without bypass was not the most restrictive transient with regard to setting the operating limit MCPR since in both the GESSAR-238 and GESSAR-251 BWR/6 analyses, this event is most limiting.

12. The applicant must provide his analysis of abnormal operational occurrences using the ODYN code.

13. The applicant must provide additional discussion of the shaft seizure accident to verify that the criteria for infrequent events are satisfied when only credit for safety grade systems is assumed.

14. ATWS procedures are to be submitted by the applicant for staff review and approval.

TMI-2 Requirements (22.0)

15. ~~We are waiting for additional information for the following items:~~

    ~~(1) II.K.1.5~~

    ~~(2) II.K.1.10~~

15 TRANSIENT AND ACCIDENT ANALYSIS

Introduction

Two groups of design basis events are evaluated in this section: anticipated operational occurrences and accidents. In order for the analysis of events in either group to be acceptable, it is required that a conservative model of the reactor be used, and that all appropriate systems whose operation (or postulated misoperation) would affect the event be included. Anticipated operational occurrences are expected to occur during the life of the power plant and are analyzed to assure that they will not cause damage to either the fuel or to the reactor coolant pressure boundary and to assure that the radiological dose is maintained within Title 10, Code of Federal Regulations, Part 20 (10 CFR 20) guidelines. Design basis accidents are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are analyzed to determine the extent of fuel damage expected and to assure that reactor coolant pressure boundary damage, beyond that assumed initially by the design basis accident, will not occur, and that the radiological dose is maintained within 10 CFR 100 guidelines.

For loss-of-coolant accidents, the acceptance criteria for the emergency core cooling system specified in Title 10, Code of Federal Regulations, Part 50.46 are: (1) the peak cladding temperature must remain below 2200 degrees Fahrenheit; (2) maximum cladding oxidation must nowhere exceed 17 percent of the total cladding thickness before oxidation; (3) total hydrogen generation must not exceed one percent of the hypothetical amount that would be generated if all the metal in the cladding cylinders, excluding the cladding surrounding the plenum volume, were to react; (4) the core must be maintained in a coolable geometry; (5) calculated core temperatures after successful initial operation of the emergency core cooling system shall be maintained acceptably low and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

15.1 Anticipated Operational Occurrences

Anticipated operational occurrences are those transients resulting from single equipment failures or single operator errors that might be expected to occur during normal or planned modes of plant operation. The acceptance criteria for these transients are based on General Design Criteria 10, 15 and 20. Criterion 10 specifies that the reactor core and associated control and instrumentation systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Criterion 15 specifies that sufficient margin shall be included to assure that design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Criterion 20 specifies that a protection system be provided that initiates automatically appropriate systems to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation including anticipated operational occurrences. Specific acceptance criteria (Standard Review Plan) for the moderate frequency transients are:

(1) Pressures in the reactor coolant and main steam systems should be maintained below 110% of the design values according to ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Overpressure Protection." For Perry, which has a design pressure of 1250 pounds per square inch gauge, the pressure should not exceed 1375 pounds per square inch gauge during any anticipated operational occurrence.

(2) Fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during any conditions of normal operation, including the effects of anticipated operational occurrences. For boiling water reactors, the minimum value of the critical power ratio reached during the transient should be such that 99.9% of the fuel rods in the core would not be expected to experience boiling transition during core wide transients. This limiting value of the minimum critical power ratio, called the

safety limit, will vary for different plants and/or product lines. For Perry, the value is 1.06 (see Section 4.4).

(3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

The applicant used the following conservative assumptions with respect to initial power, scram reactivity, reactivity coefficients, and power profiles in the analyses:

(1) Initial power 104.4 percent of rated power (corresponding steam flow rate 105 percent of rated steam flow rate).

(2) Scram reactivity characteristics accounting for end-of-cycle conditions which result in the most conservative effects. The slowest allowable control rod scram motion is assumed with a scram worth shape for all the control rods fully withdrawn at the end-of-cycle.

(3) Scram reactivity calculations incorporating a 20 percent safety conservatism factor (i.e., rod worth of 0.80 x end-of-cycle rod worth).

This includes the effect of a stuck control rod as required by General Design Criterion 26.

(4) Core burnup selected to yield the most limiting expected combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

The transients analyzed involved the following reactor scrams:

(1) Reactor vessel high pressure,

(2) Reactor vessel low water level,

(3) Turbine stop valve closure,

(4) Turbine control valve fast closure,

(5) Main steam line isolation valve closure,

(6) Neutron monitoring system scram.

Time delays to trip for each scram signal were included in the analyses.

The transient events were categorized in terms of the following system parameter variations:

1. Decrease in Core Coolant Temperature

   Transients analyzed in this group included loss of feedwater heating, feedwater control failure to maximum demand, pressure regulator failure in the open direction, inadvertent opening of a safety/relief valve and inadvertent residual heat removal shutdown cooling operation at reduced pressure and temperature conditions of the reactor.

2. Increase in Reactor Pressure

   Transients in this group included generator load rejection and turbine trip with and without turbine bypass, inadvertent MSIV closure, loss of condenser vacuum, loss of auxiliary power transformer, loss of all grid connections and loss of all feedwater flow. Failure of residual heat removal shutdown cooling was considered for conditions of reduced pressure and temperature of the reactor.

3. Decrease in Reactor Coolant System Flow Rate

   Transients in this group included trip of one or both recirculation pump motors and recirculation flow control failure to decrease flow.

4. Reactivity and Power Distribution Anomalies

Transients in this group included rod withdrawal error at power, abnormal startup of an idle recirculation loop and recirculation flow controller failure with increasing flow.

5. Increase in Reactor Coolant Inventory

The transient analyzed was inadvertent startup of the high pressure core spray pump (feedwater flow control failure to maximum demand was covered in category 1).

6. Decrease in Reactor Coolant Inventory

Inadvertent opening of a safety relief valve (covered in category 1).

The analysis of continuous control rod withdrawal during power operation was made with the three-dimensional BWR Simulator code described in NEDO-20953, "Three-Dimensional BWR Core Simulator," which has been approved by NRC. The analyses of the other abnormal operational transients were performed using the REDY computer code described in General Electric Topical Report NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor."

The transient resulting in the highest system pressure was a generator load rejection without turbine bypass which resulted in a peak system pressure about 144 pounds per square inch below the allowable maximum pressure of 1375 pounds per square inch gauge. Note, however, that the worst transient for the American Society of Mechanical Engineers overpressure protection report is an inadvertent main steam isolation valve closure with postulated failure of the position switch scram (see Section 5.2.2).

For transients categorized under Decrease in Core Coolant Temperature, the most severe transient is the loss of feedwater heating event with manual flow control. The resultant minimum critical power ratio reached is 1.08 and the peak vessel pressure is 274 pounds per square inch below the ASME code limit.

For this transient, the applicants assumed a 100 degree Fahrenheit drop in feedwater temperature. The applicant has assumed operation of a thermal power monitor (TPM) circuitry to initiate scram for this event. TPM is the primary protection system trip in mitigating the consequences of this event. TPM is a safety grade system and is designed to be single failure proof. TPM conservatively estimates thermal power by passing the APRM signal through a time constant. Surveillance testing of the TPM will be included in the technical specifications.

Inadvertent safety/relief valve opening causes a decrease in reactor coolant inventory and results in a mild depressurization event which has only a slight effect on fuel thermal margins. Changes in surface heat flux are calculated to be negligible indicating an insignificant change in minimum critical power ratio. Thus, the transient is found to be acceptable. The effect of inadvertent safety/relief valve opening on suppression pool temperature is treated in Section 6.2.

The applicants were asked to justify that operation with partial feedwater heating to extend the cycle beyond the normal end of cycle condition would not result in a more limiting change in minimum critical power ratio than that obtained using the assumption of normal feedwater heating. We require that analyses be provided prior to operation in the mode if a decision is made to operate in this mode. Until such analyses are provided we will condition the license from operation in this mode.

For transients categorized under Decrease in Reactor Coolant System Flow Rate, the most severe transient is that resulting from simultaneous trip of both recirculation pump motors. As the pumps coast down, the core void fraction increases, causing level swell in the reactor vessel and a decrease in neutron flux. Turbine trip occurs because of high water level. The minimum critical power ratio remains well above the safety limit and there is a small increase in reactor pressure.

Increased recirculation flow because of flow control failure or startup of an idle recirculation pump would result in reactivity anomalies. These events are not limiting transients and neither primary pressure boundary nor fuel damage criteria are exceeded. The transient which could cause unplanned addition to coolant inventory is the inadvertent actuation of the high pressure core spray system. The high pressure core spray system actuation has a small effect, because its flow is small compared to the recirculation flow. The transient has little effect on fuel thermal margins and on reactor system pressure.

Of the transients listed under Increase in Reactor Pressure, the limiting transients are pressure controller down scale failure and generator load rejection without bypass which result in minimum critical power ratios of 1.13 and 1.10, respectively both above the MCPR safety limiting and therefore acceptable. The applicants have proposed however, that the generator load rejection without bypass and the turbine trip without bypass transients be classified as infrequent events. The reclassification of these events has been under review by the staff and has not been approved. We will require that the applicant treat these events as being in the moderate frequency category.

We also requested additional information as to why the generator load rejection without bypass event was not the most restrictive transient with regard to setting the operating limit minimum critical power ratio since in both the GESSAR-238 and GESSAR-251 BWR/6 analyses. this event is most limiting. We will report on this item in a supplement to this report.

In analyzing anticipated operational transients, the applicants have taken credit for plant operating equipment which is not normally reviewed by us because it is not considered essential for safety. We have discussed the application of this equipment generically with General Electric. Based on these discussions, it is our understanding that the most limiting transient, aside from generator trip without bypass, that takes credit for this equipment is the excess feedwater event. Further, it is our understanding that the only plant operating equipment that plays a significant role in mitigating this event (excess feedwater) is the turbine bypass system and the level 8 high water level trip (closes turbine stop valves). In order to assure an acceptable level of performance for Perry, our position is that this equipment be identified in the plant Technical Specifications with regard to availability, setpoints, and surveillance testing. The results of our review of the Technical Specifications with respect to the level 8 trip and turbine bypass system will be addressed in Chapter 16.0.

As noted above, the operating minimum critical power ratio was also based in part on the REDY model described in NEDO-10802. During the staff review of NEDO-10802, three turbine trip tests were performed at Peach Bottom 2. As discussed in Section 4.4, these tests indicated that the results obtained with the REDY model are nonconservative for some events. We therefore require that the ODYN model, which has been approved by the staff, be used to analyze the following transients.

(1) For Thermal Limit Evaluation

(a) Feedwater controller failure - maximum demand,
(b) Generator load rejection without bypass,
(c) Turbine trip without bypass, and
(d) Main steam isolation valve closures.

(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).

We will review the results of these calculations to verify the acceptability of the operating minimum critical power ratio.

15.2 Accidents

The applicants analyzed a pump shaft seizure and pump shaft break accident. Reactor scram is sufficient to preclude violating the safety limit minimum critical power ratio (1.06) and, therefore, no fuel damage occurs. The reactor vessel pressure is maintained below the specified limit throughout the event. However, the analyses included the use of non-safety grade equipment, in particular, the turbine bypass system. We require that the applicant provide confirmation that precluding the use of non-safety grade equipment during these accidents will not result in violation of the safety limit minimum critical power ratio.

15.2.1 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 Transients Without Scram for Light Water Reactors" was issued describing the proposed type of plant modifications we believe are necessary to reduce 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 through rulemaking the required modifications to resolve anticipated transient without scram concerns as well as the required schedule for implementation of such modifications. Perry 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, in order 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 procedures 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, 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, 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 lower power levels where the existing standby liquid control system capability is sufficient to limit the pool temperature rise to an acceptable level.

We require the applicant to submit the ATWS procedures for staff review and approval. The acceptability of these procedures will be reported in a supplement to this SER.

~~15.0 SAFETY ANALYSES~~

15.4 Rod Withdrawal Events

15.4.1 Inadvertent Rod Withdrawal at Low Power

Discussion

The applicant has examined the design of the rod control system to ascertain if a single failure can lead to the uncontrolled withdrawal of a control rod during refueling and during startup and low power operation. During refueling operations interlocks ensure that all control rods are inserted while fuel is being handled over the core. When no fuel is being handled, a maximum of one rod may be withdrawn. However, the control system is designed (see Section 4.3.2) so that the core is subcritical with the highest worth rod withdrawn. Finally, the removal (from the top) of a control rod is not physically possible without removing the four fuel assemblies which surround the rod.

The uncontrolled withdrawal of a rod during reactor startup is prevented by the rod pattern control system function of the RCIS. This system is single-failure proof and enforces the banked position withdrawal sequence. Thus rod withdrawals other than those permitted in normal operation will be precluded.

Evaluation Findings

The possibilities for single failures of the reactor control system which could result in the uncontrolled withdrawal of control rods under low power startup conditions have been reviewed. The RCIS prevents the withdrawal of control rods in any but the design sequence. Thus the consequences of any rod withdrawal are within the bounds of those for normal operation.

The staff, therefore, finds that GDC 20, which requires that protection be automatically initiated, and GDC 25, which requires that a single failure of the protection system does not result in violation of specified fuel design limits, have been satisfied.

15.4.2 Rod Withdrawal Error at Power

Discussion

The "at power" rod withdrawal error transient occurs when the operator makes a procedural error and withdraws a rod continuously. While procedures call for moving rods in small increments (a few notches) in the power range and prescribes the sequence of such withdrawals, no hardware enforces this sequence. Instead, the rod withdrawal limiter mode of the RCIS acts to limit the amount by which a control rod or control rod gang may be moved in a single action.

A statistical generic analysis of this event has been performed for BWR-6 reactors and is reported in Appendix 15B of the GESSAR II. It is also reported in Appendix 15B of the Perry FSAR.

For the statistical analysis, anticipated operating states (including rod patterns) are developed for several cycles of a BWR-6. These states are examined to establish the anticipated MCPR values as a function of core power. A Technical Specification curve of allowable initial MCPR (IMCPR) as a function of power is then defined so that there is margin between this limit and the expected plant performance. The change in MCPR (ΔMCPR) is then calculated for ganged rod withdrawal events which are initiated from reactor states (rod patterns) used to establish the IMCPR curve (analyses have shown that ganged withdrawals are always more limiting). The ΔMCPR values are then expressed as a function of withdrawal distance and core power, and withdrawal distances are determined as a function of initial power, such that there is a 95 percent probability with 95 percent confidence that the safety limit MCPR for the core will not be violated. The curve of withdrawal limit as a function of initial power is then divided into two segments--above 70 percent of full power and between 20 and 70 percent of full power--and constant values of the withdrawal distance defined for each segment. The rod withdrawal limiter is then designed to enforce these limits which are 12 in. above 70 percent of full power and 24 in. between 20 and 70 percent of full power. Below 20 percent of full power, rod motions are constrained by the rod pattern control system mode of the RCIS.

The staff has reviewed the analysis of the rod withdrawal event at power and conclude that it is acceptable. This conclusion is based on the following:

(1) The BWR simulator code, which has been reviewed and approved by the staff, was used to perform the analyses.

(2) The data base used for the analyses covered the range of BWR-6 sizes and exposure cycles.

(3) The allowed withdrawal distances are conservative with respect to the calculated results.

(4) The Technical Specification values of the permitted initial MCPR as a function of core power are conservative with respect to the calculated values.

It should be noted that this analysis is not applicable to a control cell loading strategy or high-energy/high-discharge exposure cycles. Use of such cycles in the Perry reactor will require that compliance checks be performed and documented to demonstrate the applicability of the generic analysis.

Evaluation Findings

The possibilities for single failures of the reactor control system which could result in uncontrolled withdrawal of control rods beyond normal limits under power operation conditions have been reviewed. The scope of the review has included investigations of possible initial conditions and the range of reactivity insertions, the course of the resulting transient and the instrumentation response to the transient. The methods used to determine the peak fuel rod response, and the input into that analysis, such as power distributions, rod reactivities, and reactivity feedback effects of moderator and fuel temperature changes, have been examined.

The staff concludes that the requirements of GDC 10, 20, and 25 have been met. This conclusion is based on the fact that the applicant has met the requirements of GDC 10 that the specificed acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control systems will not cause the specified acceptable fuel design limits to be exceeded. These requirements have been met by comparing the resulting extreme operating conditions and response for the fuel (fuel duty) with the acceptance criteria for fuel damage (for examples, critical heat flux, fuel temperatures, and clad strain limits should not be exceeded) to ensure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analysis of maximum transients for single error control rod malfunctions have been confirmed, that the analytical methods and input data are reasonably conservative, and that specified acceptable fuel design limits will not be exceeded.

15.4.7 Operation of a Fuel Assembly in an Improper Position - Fuel Misloading Event

Discussion

Two sorts of fuel misloading events may be considered--misorienation of a fuel assembly in its proper location, and loading a fuel assembly into an improper location. The first of these events has trivial consequences for a C-lattice in the first cycle because the assembly fuel design is symmetric with respect to rotation. The slight tilt in the assembly caused by the misorientation has a negligible effect on the thermal-hydraulic performance of the fuel in the first cycle and tends to improve that performance in succeeding cycles.

The initial core consists of three bundle types with average enrichments that are high, medium, or low with correspondingly different gadolinia concentrations. The fuel bundle loading error involves interchanging a bundle of one enrichment with another bundle of a different enrichment. The following fuel loading errors can be conceived for an initial core:

(1) A high-enriched bundle is misloaded into a low-enriched bundle location.

(2) A medium-enriched bundle is misloaded into a low-enriched bundle location.

(3) A low-enriched bundle is misloaded into a high-enriched bundle location.

(4) A low-enriched bundle is misloaded into a medium-enriched bundle location.

(5) A medium-enriched bundle is misloaded into a high-enriched bundle location.

(6) A high-enriched bundle is misloaded into a medium-enriched bundle location.

Because all low-enriched bundles are located on the core periphery, the two possible fuel loading errors consisting of the misloading of high- or medium-enriched bundles into a low-enriched bundle location (Types 1 and 2 from above) are not significant. In these cases, the higher reactivity bundles are moved to a region of lower importance resulting in an overall improvement in performance.

Types 3, 4, and 5 make up a class of events which consist of misloading a bundle into a location intended for a higher enrichment. In this case if the lower enrichment bundle is loaded next to an LPRM string the detector response will be different from that in a properly loaded four-bundle array. Two effects are present--a reduction in detector response due to the low enrichment bundle and an increase due to the softened neutron spectrum resulting from the decreased thermal absorption present (due to the lower enrichment). The latter effect dominates and the detector reads too high--a conservative direction.

A bounding misloading event analysis was performed for the Perry reactor by neglecting the spectral softening effect for the Type 3 event. It is assumed that the four bundle array containing the misloaded bundle is put on operating limits. Because of the under-response of the LPRM string, the mirror image bundles (which are properly loaded) will exceed operating limits. A change of 0.11 in CPR is calculated for this event along with a change of 0.9 kilowatts per foot in LHGR.

If the spectral softening effect is taken into account, the Type 6 event is limiting. For this event--loading a high enrichment bundle into a medium enrichment location--the initial effect is small since there are different amounts of gadolinium in two bundles such that their reactivities are similar. However, as the gadolinium burns out, the high enrichment bundle becomes increasingly more reactive than the medium enrichment bundle. If, at the end of cycle, one of the mirror four-bundle arrays (assumed to be instrumented) is placed on operating limits, the array containing the misloaded bundle will exceed limits.

The consequences of this type of event have been analyzed for a similarly loaded BWR/6. The results show a change in CPR of 0.10 and a change in LHGR rate of 1.4 kilowatts per foot.

The analyses were performed with the BWR Simulator code which has been reviewed and approved by the staff. In all cases, the changes in thermal parameters (CPR and LHGR) are well within the operating margins for these quantities. We conclude that the analysis of the fuel misloading event is acceptable.

Evaluation Findings

The staff has evaluated the consequences of a spectrum of postulated fuel loading errors and concludes that the analyses provided by the applicant have shown for each case considered that either the error is detectable by the available instrumentation (and hence remediable) or the error is undetectable but the offsite consequences of any fuel rod failures are a small fraction of 10 CFR Part 100 guidelines. The applicant affirms that the available instrument will be used before the start of a fuel cycle to search for fuel loading errors.

The staff concludes that the applicant has met the requirements of GDC 13 with respect to providing adequate provisions to minimize the potential of a mis-loaded fuel assembly going undetected and meets 10 CFR 100 with respect to mitigating the consequences of reactor operations with a misloaded fuel assembly. These requirements have been met by providing acceptable procedures and design features that will minimize the likelihood of loading fuel in a location other than its designated place.

15.4.9 The Rod Drop Accident

Discussion

The postulated rod drop accident occurs when a rod which has been stuck in the upper portion of the core becomes disconnected from the rod drive, the drive is subsequently withdrawn, and the rod becomes unstuck and falls rapidly onto the drive. This results in a power excursion which could, under certain circumstances, result in local fuel damage.

The consequences of the rod drop accident depend chiefly on the amount of reactivity inserted into the core by the dropping rod and on the initial thermal hydraulic conditions of the core. Dependence on rod drop speed, Doppler feedback coefficient, the shape of the scram curve, and the scram speed are less pronounced. The analysis of the rod drop accident has been performed on a generic basis by the General Electric Company and is reported in NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," and Supplements 1 and 2 to that report. The calculation is performed under the following conservative assumptions:

(1) No thermal-hydraulic feedback is assumed.

(2) The least negative Doppler coefficient which is anticipated is used.

(3) The rod drop speed is assumed to be that measured for the rod design plus three standard deviations.

(4) The scram speed is the Technical Specification value.

(5) The shape of the scram curve is assumed to be that which starts with all rods out of the core. This configuration results in the longest delay before significant reactivity is inserted into the core.

In addition, the calculational model contains conservatism. For example, the axial flux shape is assumed to remain constant throughout the excursion. This means that the energy deposition in the hot pellet is maximized.

The enthalpy rise in the hot pellet is plotted as a function of the worth of the dropped rod in NEDO-10527, Addendum 1. For the design calculation described above, a rod worth of approximately 1.4 percent reactivity change is required to produce an enthalpy rise of 280 cal/gm, which is the staff acceptance criterion.

To assess the extent of the conservatism in the assumption of no thermal-hydraulic feedback in the design calculations, the staff consultant, Brookhaven National Laboratories, performed a series of calculations which included this effect. The results are reported in BNL-NUREG-28109, "Thermal-Hydraulic Effects on Center Rod Drop Accidents in a Boiling Water Reactor," July 1980. These results show that if thermal-hydraulic feedback is included in the calculations, the resulting enthalpy rise is less than 140 cal/gm for a rod worth of 1.4 percent reactivity change. Thus it may be concluded that a large conservatism factor exists in the design calculations.

The Perry reactor has a rod pattern control system which enforces the banked position withdrawal sequence. This sequence has been described in NEDO-21231, "Banked Position Withdrawal Sequence," September 1976. This document shows that the maximum worth of a potential dropped rod during withdrawal of the first 50 percent of the rods is 0.62 percent reactivity change during the first cycle and is less during succeeding cycles.

The maximum worth potential dropped rod during withdrawal of the second 50 percent of the rods is 0.75 percent reactivity change during the first cycle and 0.83 percent reactivity change during the equilibrium cycle. Using the largest of these values, the design calculation would predict an enthalpy rise of 135 cal/gm. The calculation with thermal-hydraulic feedback would predict less than 75 cal/gm.

In view of the low value of maximum fuel enthalpy cladding failure would not be expected. However, for purposes of evaluating environmental consequences, it is conservatively assumed that 770 rods suffer cladding failure. Similarly, the low value of fuel enthalpy would result in an insignficant pressure pulse in the core.

Evaluation Findings

The staff concludes that the analysis of the rod drop accident is acceptable The applicant met the requirements of GDC 28 with respect to preventing postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding or cause sufficient damage that would significantly impair the capability to cool the core. The staff has evaluated the applicant's analysis of the assumed control rod drop accident and finds the assumptions, calculational techniques, and consequences acceptable. Because the calculations predict peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. In view of the low fuel enthalpy for the maximum with rod no significant pressure pulse is anticipated. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.

15.X

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

In order to demonstrate the adequacy of the plant's engineered safety features, the applicant calculated the offsite consequences which could result from the occurrence of each of several design basis accidents and presented the results of these computations in the FSAR.

We have independently performed similar calculations in order to confirm the effectiveness of the engineered safety features. The staff's computed doses for these potential accidents are listed in Table 15-1. Where these doses are significantly dependent upon plant conditions at the time of the accident, or arise through multiple leakage paths, separate doses for extreme conditions or separate dose contributions from the individual leakage paths have been listed. The computed doses in Table 15-1 are in a form which allows direct comparison with the dose guidelines of 10 CFR Part 100.11, and are expressed as rems of thyroid and whole body exposure.

The procedures used in our review were consistent with the applicable Standard Review Plan Sections and

Regulatory Guides. The values of the atmospheric diffusion parameters (X/Q's) in all of the computations discussed below are those presented in Section 2.3.4 of this report.

Other assumptions are discussed in the following sections and are outlined in Tables 15-2 through 15-4.

15.X.1 LOSS-OF-COOLANT ACCIDENT (RADIOLOGICAL CONSEQUENCES)

A design basis loss-of-coolant accident was postulated for the Perry Nuclear Power Station using the methods described in the Appendices to Standard Review Plan 15.6.5 and in Regulatory Guide 1.3. To mitigate the consequences of this accident, the plant design includes two major active systems: the annulus exhaust gas treatment system (AEGTS) and the main steam isolation valve leakage collection system (LCS), both discussed earlier in Section 6.5. The AEGTS filters and exhausts air from the secondary containment to prevent diffusive loss of airborne iodine from the plant, while the LCS prevents fission product gases from leaking into the turbine building via the main steam lines. Together, these systems assure that any leakage to the secondary containment passes to the environment only through the AEGTS. In addition to the leakage from the secondary containment, however, the possibility exists of leakage from the primary containment directly to the environment. Such unfiltered leakage to the environment is termed "bypass".

Computed potential thyroid doses are predominately dependent upon the assumed distribution of containment leakage between the bypass and ESF-filtered pathways. The computed doses for this accident are listed in Table 15-1, and the assumptions used in these computations are outlined in Table 15-2.

The doses computed for this accident are less than the guideline values of 10 CFR Part 100.11, and we conclude that the Perry Nuclear Power Station is adequately designed to mitigate the offsite consequences arising from a loss-of-coolant accident.

15.X.2 CONTROL ROD DROP ACCIDENT

In this postulated event, the highest worth control rod becomes decoupled from its drive mechanism at a fully inserted position in the core. The drive mechanism is withdrawn, but the decoupled control rod is assumed to be stuck in place. At a later moment, the control rod suddenly falls free and drops out of the core. This results in the insertion of a large positive reactivity into the core and results in a localized power excursion. The termination of this excursion is accomplished by automatic safety features, and no operator action is required.

The Perry plant has a rod control and information system (RCIS) that will limit the worth of any control rod by regulating the withdrawal sequence. The RCIS uses redundant input to provide assurance of control rod position. The applicant has estimated that such a rod drop would result in the failure of the cladding of 770 fuel rods with 6 rods melted.

In evaluating the radiological consequences, we have used the applicant's estimate for failed fuel rods, with the amount of activity accumulated in the fuel clad gap as given in Regulatory Guide 1.77. The failed fuel rods release fission products to the primary coolant and ultimately to the environment. The assumptions used for this analysis are listed in Table 15-3. The calculated doses listed in Table 15-1, are well within the dose guideline values of 10 CFR Part 100. Based on our review, we conclude that the Perry Nuclear Power Plant is effectively designed to control the release of fission products following a postulated control rod drop accident.

15.X.3 FUEL HANDLING ACCIDENT

A postulated fuel handling accident was evaluated using assumptions consistent with position C.1.a through C.1.k of Regulatory Guide 1.25. The kinetic energy of a single falling fuel assembly was assumed to break open the maximum possible number of fuel rods using perfect mechanical efficiency. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods was assumed to occur, followed by the release of these activities through the pool water. Radiation monitors located within the normal ventilation system are designed to shut down that system and activate the FHAES, such that no significant fraction of the release could escape untreated. The FHAES has been described earlier in section 6.5, and this system would automatically respond to a radioactivity release from a fuel handling accident.

A list of the assumptions obtained for Perry by the application of Regulatory Guide 1.25 positions is contained in Table 15-4. The amount of radioiodine described in position C.1.d and used here is extremely conservative, and is several times larger than that permitted by the vapor pressure of elemental iodine. The offsite doses computed using these assumptions are listed in Table 15-1, and are well within the guideline values of 10 CFR Part 100. The fuel handling and storage provisions also meet the applicable portion of General Design Criterion 61, (i.e., "(3) with appropriate containment, confinement, and filtering systems,") The staff, therefore, concludes that the plant features designed to mitigate the consequences of fuel handling accidents are adequate.

In the staff's review of the spent fuel cask drop accident, it was noted that the potential vertical drop distances onto hard surfaces were less than 30 feet and that the cask is not permitted to travel over spent fuel or safety-related equipment. Because the most severe drop is within the designed strengths of the barriers to fission product release, the radiological consequences of a spent fuel cask drop accident have not been evaluated (as specified in Standard Review Plan 15.7.5).

15.X.4 STEAM LINE BREAK ACCIDENT

One of the four main steam lines was postulated to rupture between the outer isolation valve and the turbine control valves. The applicant has analyzed this hypothetical accident and has concluded that no more than 64,300 kilograms of steam and reactor coolant would be lost through the break prior to its automatic isolation. This loss is sufficiently large to conservatively encompass the coolant release which might occur during any steam line failure outside containment.

Our evaluation of this accident followed the procedures outlined in Standard Review Plan 15.6.4, "Radiological Consequences of Main Steam Line Failure Outside Containment," and in Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors." We assumed that all water and steam lost through the break contained the same concentration of radioactive isotopes as the reactor coolant, and that all of this radioactivity was released directly to the environment as an airborne contaminant.

Two reactor coolant conditions were assumed for the evaluation. In Case 1, the coolant was assumed to be contaminated to the limit allowed by the Standard Technical Specifications for boiling water reactors during normal operation, which is a concentration of 0.2 microcuries per gram of dose equivalent I-131. In Case 2, a concentration of 4.0 microcuries per gram of dose equivalent I-131 was assumed (this being the Standard Technical Specification limit above which the reactor would be required to be shutdown). The Standard Review Plan acceptance criteria are less than the guideline values of 10 CFR Part 100.11 for Case 2, and less than 10% of these values for Case 1. Dose equivalent I-131 is defined as a mixture of five iodine isotopes yielding the same inhalation dose as the stated amount of pure I-131. We have also considered the amounts of 13 noble gas isotopes which would also be expected to be dissolved in the coolant due to the same fuel leakage processes which would lead to iodine contamination.

The calculated doses are listed in Table 15-1, and are less than the appropriate acceptance criteria. Therefore, we conclude that the dose c nsequences of this accident are sufficiently mitigated by limits upon the reactor coolant activity and isolation valve closure time, and that the design of the plant with respect to this accident is acceptable.

15.X.5 SMALL LINE FAILURE

General Design Criterion 55 specifies provisions to assure isolation of all pipes carrying reactor coolant which penetrate the containment building. Exempted from these specifications are small diameter pipes which must be continuously connected to the primary coolant system in order to perform necessary functions. For these lines, generally called instrument lines, methods of mitigating the consequences of a rupture are necessary due to the lack of isolation capability.

The instrument lines at Perry contain 1/4 inch diameter flow restriction orifices to limit the loss of coolant during the time needed to respond to an instrument line rupture by shutting down and depressurizing the reactor. All instrument lines at Perry which penetrate containment terminate in spaces processed by ESF filters. Thus, with respect to instrument line breaks at Perry, the design is equivalent to that of the recently reviewed Clinton Power Station where the application of conservative assumptions showed that the doses are small fractions of the 10 CFR Part 100 exposure guideline values. Thus, we find that the Perry design is acceptable with respect to this accident.

TABLE 15-1

RADIOLOGICAL CONSEQUENCES OF SELECTED DESIGN BASIS ACCIDENTS

	0-2 hour Doses, Exclusion Area Boundary, Rem		0-30 Day Doses, Low Population Zone, Rem	
POSTULATED ACCIDENT	Thyroid	Whole Body	Thyroid	Whole Body
Loss of Coolant				
Total	120	9.0	152	3.1
Steam Line Break				
Case 1	4.4	<0.1	0.54	<0.1
Case 2	89	<0.2	11	<0.1
Control Rod Drop	1.3	0.13	0.9	<0.1
Fuel Handling				
Inside containment	6.9	0.94	2.4	0.12
Outside containment	7.2	1.0	0.9	0.12
Loss-of-Coolant Accident Doses during intervals within first 30 days at Low Population Zone				
0-2 hours			15	1.1
2-8 hours			40	1.2
8-24 hours			28	0.5
1-4 days			42	0.2
4-30 days			27	0.1

TABLE 15-1 (Continued)

RADIOLOGICAL CONSEQUENCES OF SELECTED DESIGN BASIS ACCIDENTS

Atmospheric Diffusion Parameters	
Exclusion area boundary:	
0-2 hours	6.7×10^{-4} seconds/cubic meter
Low population zone boundary:	
0-8 hours	8.2×10^{-5} seconds/cubic meter
8-24 hours	5.2×10^{-5} seconds/cubic meter
1-4 days	1.9×10^{-5} seconds/cubic meter
4-30 days	4.7×10^{-6} seconds/cubic meter
Minimum Exclusion Area Boundary Distance	863 meters
Low Population Zone Distance	4002 meters

TABLE 15-2

ASSUMPTIONS USED IN COMPUTING LOSS OF COOLANT ACCIDENT DOSES

Power level	3834 Mwt, maximum inventory
Free Containment Volumes	
Primary	1.5×10^6 cubic feet
Secondary	4.5×10^5 cubic feet
Primary Containment Leak Rate	0.20%/day
By-pass Fraction	4%
Main Steam Line Isolation Valve Leak Rate (4 lines)	1.67 standard cubic feet/min (25 standard cubic feet/hr/valve)
Annulus Exhaust Gas Treatment System	
iodine filter efficiency	99% (all iodine species)
flow rate	1950 cubic feet/minute
Initial Fractions of Core Inventory Available for Leakage	
iodine	25%
noble gas	100%

TABLE 15-3

ASSUMPTIONS USED IN COMPUTING ROD DROP ACCIDENT DOSES

Power Level (Mwt)	3834
Peaking Factor	1.55
Number of Fuel Rods Perforated	770
Number of Fuel Rods Melted	6
Total Number of Rods in Core	46,376
Condenser Leak Rate (%/day)	1.0
Fraction of Fission Product Inventory Release to Coolant	
Iodines	50%
released to condenser	10%
available for release after plate-out and partitioning	10%
Noble Gases	100%
released to condenser	100%

<u>Atmospheric diffusion values</u>

0-2 hour, Exclusion Boundary	6.7×10^{-4}
0-8 hour, Low Population Zone	8.2×10^{-5}

TABLE 15-4

ASSUMPTIONS USED IN COMPUTING FUEL HANDLING ACCIDENT DOSES

Power level (Mwt)	3834
Peaking factor	1.55
Number of fuel rods damaged	108
Total number of rods in core	46,376
Filter Efficiencies	
Inside containment	
Organic	99%
Elemental	99%
Outside containment	
Organic	95%
Elemental	95%
Shutdown Time (hours)	24
Inventory released from damaged rods	
Iodine and Noble Gases	10%
Kr 85	35%
Iodine fraction	
Organic	0.25
Elemental	0.75
Atmospheric diffusion values	
0-2 hour, Exclusion Boundary	6.7×10^{-4}
0-8 hour, Low Population Zone	8.2×10^{-5}

SECTION 16

TECHNICAL SPECIFICATIONS

QUALITY ASSURANCE

SECTION 17

TMI-2 REQUIREMENTS

SECTION 22

OPEN AREAS
CONTAINMENT SYSTEMS

TMI -2 Requirements (22.0)

15. We are waiting for additional information for the following items: XXXXXXXX

(1) II.K.1.5
(2) II.K.1.10
(3) II.K.3.3
(4) II.K.3.16
(5) II.K.3.17
(6) II.K.3.18
(7) II.K.3.21
(8) II.K.3.25
(9) II.K.3.30
(10) II.K.3.31
(11) II.K.3.45

Items 2,3,4 and 9 are confirmatory. We have agreed with the applicant's proposed plan for resolving these issues. We require documentation to close them.

The applicant has not yet provided the information we require regarding the TMI Action Items, including the measures for hydrogen control in the event of a degraded core accident.

Re: Section 6.2.7 (not addressed by applicant)

II.E.4.1, Dedicated Hydrogen Penetration
II.E.1, Attach. 4 - Containment Pressure Monitor
II.F.1 Attach 5 - Containment Water Level Monitor
II.F.1 Attach 6 - Containment Hydrogen Monitor
II.B.8 Rulemaking Proceeding on Degrade Core Accidents

22.0 TMI-2 REQUIREMENTS

II.B.1 Reactor Coolant System Vents

Requirements: See Item II.B.1 of NUREG-0737, November 1980

Discussion and Conclusions

In a letter dated November 5, 1981 from Dalwyn R. Davidson (CEI) to R. L. Tedesco (NRC), the applicant has stated that the primary venting capability is provided by the 19 power operated safety/relief valves. Each of the safety relief valves is seismically and Class 1E qualified and the air supply to the eight 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. Positive position indication for each valve will be provided in the control room as discussed under item II.D.3 of this chapter. Additional venting capability is provided via a reactor vessel head vent valve and through operation of the turbine driven reactor core isolation cooling system. No additional accident analyses have been provided as a result of a break in any of these vent lines because a more bounding complete steam line break is part of the plant's design basis. This is acceptable to us.

II.B.2 DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POSTACCIDENT OPERATIONS

Position

With the assumption of a postaccident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% 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 postaccident operations of these systems.

Each licensee shall provide for adequate access to vital areas and 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 through the facility.

Discussion and Conclusion

In Section 12.6 of the Perry nuclear power plant FSAR, the applicant describes the radiation and shielding design review for access to areas in postaccident situations.

The source terms used by Cleveland Electric Illuminating Company (CEI) in their evaluation are those established in NUREG-0737 for liquid-containing systems and gas-containing systems.

The systems designed to function after an accident included: containment, residual heat removal system, core spray system, reactor core isolation cooling system, and postaccident sampling system. It is not clear whether the standby gas treatment system and the gaseous radwaste system are designed to function after an accident. The applicant should verify that these systems were included in the review or explain why these systems were excluded. This is an open item. The applicant has stated that other systems and areas will be isolated and are not necessary for postaccident operations.

Vital areas for the Perry nuclear power plant assumed to require postaccident personnel access include: the control room and technical support center for continuous occupancy; the remote shutdown panel for frequent occupancy; and the sampling station and sample analysis area for infrequent occupancy. These areas meet the occupancy and radiation design objectives given in Table 12.6.1 and meet the requirements of NUREG-0737 and GDC 19. Other areas are not considered vital due to remote operations capability.

Time-dependent source terms were used to determine dose rates for vital areas and vital area access routes. Tables of isotopes and gamma ray source terms and maps providing zonal, postaccident dose rates are provided.

On the basis of our review, we have concluded with the exception of clarification on whether the standby gas treatment and gaseous radwaste system were designed to function after an accident, that the applicant has performed a radiation and shielding design review for vital area access in accordance with Item II.B.2 of NUREG-0737.

Special Problems

In order to complete our review of TMI-related items, the applicant should provide information clearly indicating that the standby gas treatment system and/or the gaseous radwaste system were designed to function after an accident or explain why they were excluded.

NUREG-0737, II.B.3 - Post Accident Sampling Capability

Requirement

Provide a capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples, without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides tnat are indicators of severity of core damage (e.g., noble gases, iodines, cesiums and non volatile isotopes), hydrogen in the cont. nment atmosphere and total dissolved gases or hydrogen, boron and chloride in reactor coolant samples in accordance with the requirements of NUREG-0737.

To satisfy the requirements, the application should (1) review and modify his sampling, chemical analysis and radionuclide determination capabilities as necessary to comply with NUREG-0737, II.B.3, (2) provide the staff with information pertaining to system design, analytical capabilities and procedures in sufficient detail to demonstrate that the requirements have been met.

EVALUATION AND FINDINGS

The applicant has committed to a post-accident sampling system that meets the requirements of NUREG-0737, Item II.B.3, but has not provided the technical information required by NUREG-0737 for our evaluation. Implementation of the requirement is not necessary prior to low power operation because only small quantitites of radionuclide inventory will exist in the reactor coolant system and therefore will not affect the health and safety of the public. Prior to exceeding 5% power operation the applicant must demonstrate the capability to promptly obtain reactor coolant samples in the event of an accident in which there is core damage, consistent with the condtions stated below.

1. Demonstrate compliance with all requirements of NUREG-0737, II.B.3 for sampling, chemical and radionuclide analysis capability, under accident conditions.
2. Provide sufficient shielding to meet the requirements of GDG-19, assuming Regulatory Guide 1.3 source terms.
3. Commit to meet the sampling and analysis requirements of Regulatory Guide 1.97, Revision 2.
4. Verify that all electrically powered components associated with post accident sampling are capable of being supplied with power and operated, within thirty minutes of an accident in which there is core degradation, assuming loss of off site power.

5. Verify that valves which are not accessible for repair after an accident are environmentally qualified for the conditions in which they must operate.
6. Provide a procedure for relating radionuclide gaseous and ionic species to estimated core damage.
7. State the design or operational provisions to prevent high pressure carrier gas from entering the reactor coolant system from on line gas analysis equipment, if it is used.
8. Provide a method for verifying that reactor coolant dissolved oxygen is at <0.1 ppm if reactor coolant chlorides are determined to be >0.15 ppm.
9. Provide information on (a) testing frequency and type of testing to ensure long term operability of the post accident sampling system and (b) operator training requirements for post accident sampling.
10. Demonstrate that the reactor coolant system and suppression chamber sample locations are representative of core conditions.

In addition to the above licensing conditions the staff is conducting a generic review of accuracy and sensitivity for analytical procedures and on-line instrumentation to be used for post-accident analysis. We will require that the applicant submit data supporting the applicability of each selected analytical chemistry procedure or on-line instrument along with documentation demonstrating compliance with the licensing conditions four months prior to exceeding 5% power operation, but review and approval of these procedures will not be a condition for full power operation. In the event our generic review determines a specific procedure is unacceptable, we will require the applicant to make modifications as determined by our generic review.

The above items on post-accident sampling capability will have to be satisfactorily addressed by the applicant or we will condition the license.

II.F.1(3) CONTAINMENT HIGH-RANGE RADIATION MONITOR

Position

In containment, radiation-level monitors with a maximum range of 10^7 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.

Discussion and Conclusion

The Cleveland Electric Illuminating Company (CEI) has committed to installing four high-range gamma monitors at Perry that meet the criteria of Table II.F.1-3 of NUREG-0737 with ranges of 1 R/hr to 10^7 R/hr. CEI intends to locate two monitors in the drywell and two monitors in the reactor building. Plant layout drawings showing the location of the monitors have not been submitted to date. The applicant should provide these drawings when they are available. This is an open item.

The monitors will be powered from an independent 120-volt ac diesel-backed bus and will be provided with continuous readout and multipoint recorders in the control room. CEI intends to calibrate the monitors in accordance with the criteria outlined in Table II.F.1.3 of NUREG-0737.

On the basis of our review, we have concluded, upon receipt of plant layout drawings showing the location of the monitors, that the applicant meets our position in Item II.F.1(3) of NUREG-0737.

Special Problems

In order to complete our review of Item II.F.1(3) of NUREG-0737, the applicant should provide a plant layout drawing showing the location of the four high-range monitors.

II.K.1.22 Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System is Not Operable

REQUIREMENT: See Item 3 of Bulletin 79-08, NUREG-0560, May 1979

Discussion and Conclusions

By letter dated October 30, 1981 from Dalwyn R. Davidson (CEI) to R. L. Tedesco (NRC) the applicant forwarded his description of the actions necessary for heat removal when the main feedwater system is inoperable.

continued on next page.

The applicant stated that if the main feedwater system is not operable, a reactor scram will automatically be initiated when reactor water level falls to Level 3. The operator can remote manually initiate the reactor core isolation cooling (RCIC) system from the main control room or if the operator takes no action, reactor water level will continue to decrease from boil-off until the low-low level setpoint, Level 2, is reached. At this point, the main steam line will be isolated automatically, and the high pressure core spray (HPCS) system and the RCIC system will be automatically initiated to supply makeup water to the reactor pressure vessel. These systems will continue automatic injection until the reactor water level reaches Level 8, at which time the HPCS and RCIC systems are tripped. The high pressure core spray system (HPCS) will restart automatically once the high-level trip signal clears and a low-low level (Level 2) signal is received. The RCIC design is being modified to provide for automatic restart following an L8 trip and a subsequent low-low level signal (see Item II.K.3.13 of this chapter).

If the vessel is isolated, reactor vessel pressure is regulated by automatic or remote manual operation of the main steam relief valves which blowdown to the suppression pool. In this case, the suppression pool cooling mode of the residual heat removal system is used to transfer heat to the ultimate heat sink. This requires remote manual alignment of the residual heat removal system valves and startup from the control room of the associated service water system. Reactor vessel heat removal may also be accomplished while the vessel is isolated by operator action to align the residual heat removal system for the steam condensing mode of operation. This also involves remote valve alignments and startup of the residual heat removal service water system.

For the accident situations with the reactor vessel at high pressure, the high pressure core spray system is used 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 3 water level permissive, Level 1 water level, pressure in at least one low pressure injection system and the runout of a 120 second timer which starts with the coincidence of the other four signals.

Based on the description provided and the applicant's commitment to modify the RCIC system for automatic restart as noted above, we find the response to this item to be acceptable.

II.K.3.13 Separation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation Levels.

Requirement: See NUREG 0737, Item II.K.3.13, November, 1980.

Discussion and Conclusions

At Perry, the high pressure core spray (HPCS) and RCIC are both initiated at low-water level Level 2. Perry does not employ a high pressure coolant injection (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 results of this study were forwarded to us by letter dated December 24, 1980 from D. B. Waters (BWR Owners Group) to D. G. Eisenhut (NRC). 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 study concluded the following: (1) for rapid level changes associated with accident scenarios and severe transients, HPCS and RCIC initiation would be essentially simultaneous in that possible separation distances could not preclude HPCS challenges; (2) 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; and (3) no

*Also see Item II.K.3.18 for possible modifications to the ADS logic.

depressurization will occur* if all of the following signals are present: high drywell pressure, Level 3 water level permissive, Level 1 water level, pressure in at least one low pressure injection system and the runout of a 120 second timer which starts with the coincidence of the other four signals.

Based on the description provided and the applicant's commitment to modify the RCIC system for automatic restart as noted above, we find the response to this item to be acceptable.

II.K.3.13 Separation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation Levels

Requirement: See NUREG 0737, Item II.K.3.13, November, 1980

Discussion and Conclusions

At Perry, the high pressure core spray (HPCS) and RCIC are both initiated at low-water level Level 2. Perry does not employ a high pressure coolant injection (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 results of this study were forwarded to us by letter dated December 24, 1980 from D. B. Waters (BWR Owners Group) to D. G. Eisenhut (NRC). 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 study concluded the following: (1) for rapid level changes associated with accident scenarios and severe transients, HPCS and RCIC initiation would be essentially simultaneous in that possible separation distances could not preclude HPCS challenges; (2) 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; and (3) no

*Also see Item II.K.3.18 for possible modifications to the ADS logic.

significant reductions in thermal cycles is achievable by separating the setpoints, nor is a reduction in cycles necessary.

With regard to automatic restart of the RCIC system on low water level, the applicant has stated that this modification is being incorporated at Perry.

We conclude that for Perry, the separation of HPCS and RCIC initiation levels is unnecessary at this time, but note that the applicant is subject to the results of the ongoing generic evaluation of this topic. The applicant is committed to install the modifications four months before operating license is issued. We find this acceptable.

II.K.3.15 Modify Break Detection Logic to Prevent Spurious Isolation of High Pressure Coolant Injection and Reactor Core Isolation Cooling System

Requirement: See NUREG 0737, Item II.K.3.15, November 1980

Discussion and Conclusion

In a letter dated October 30, 1981 from Dalwyn R. Davidson (CEI) to R. L. Tedesco (NRC), the applicant identified a circuit modification to assure that trips initiated by signals from pressure instruments used to sense flow in the RCIC system are actually based on continuous high steam flow. The modification is a time delay relay which is to be added to the logic of the RCIC systems. This conceptual design is acceptable. The applicant is committed to the installation of the modification four months before operating license is issued. We find this acceptable.

II.K.3.44 Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure

Requirement: See Item II.K.3.44 of NUREG 0737, November 1980

Discussion and Conclusions

In a letter dated November 3, 1981, from Dalwyn R. Davidson to D. G. Eisenhut (NRC), the applicant endorsed as applicable to Perry the results of the Owners

Group study in this area. The Owners Group report was submitted to us by letter dated December 29, 1980 from D. B. Waters (Owners Group) to D. G. Eisenhut (NRC). The evaluation states that the worst case transient-with-single-failure combination for BWR/6 plants is the loss of feedwater event with failure of the high pressure core spray system. A stuck open relief valve was also considered in addition to the high pressure core spray failure. The results of these studies indicate that the core remains covered during the whole course of the transient either due to reactor core isolation cooling system operation, or automatic or manual depressurization permitting low pressure inventory makeup. The operator action assumed in the analysis is to manually depressurize the vessel to permit low pressure injection. Based on the results of the Owners Group Study and their applicability to Perry we find the applicants' response acceptable for this item.

III.D.3.3 IMPROVED INPLANT INSTRUMENTATION UNDER ACCIDENT CONDITIONS

Position

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.

Discussion and Conclusion

The Cleveland Electric Illuminating Company (CEI) is currently evaluating two methods of monitoring radioactive iodine during accident conditions. The first uses a silver zeolite cartridge evaluated with a single-channel analyzer. The second method uses a silver-impregnated silica gel canister. The silica gel is then measured with a GM tube inserted into the canister and the reading converted to an iodine concentration. The applicant intends to submit information on the number and type of samplers, sample media, flushing methods, and sample analysis system in October 1981. These are open items.

Special Problems

In order to complete our review of Item III.D.3.3 of NUREG-0737, the applicant should provide the following information:

1. The method of sampling.
2. The number and type of samplers.
3. Sample media to be used.
4. Sample flushing methods.
5. Sample analysis equipment, type, and location (~~most~~ must have low background area).