NUREG-0887
Supplement No. 5

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 1985

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ABSTRACT

Supplement No. 5 to the Safety Evaluation Report (NUREG-0887) on the application filed by the Cleveland Electric Illuminating Company on behalf of itself and as agent for the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company, and the Toledo Edison Company (the Central Area Power Coordination Group or CAPCO), as applicants and owners, for a license to operate the Perry Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-440 and 50-441), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Lake County, Ohio, approximately 35 miles northeast of Cleveland, Ohio. This supplement reports the status of certain issues that had not been resolved at the time of publication of the Safety Evaluation Report and Supplement Nos. 1 through 4 to that report.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ADS	automatic depressurization system
AEOD	Office of Analysis and Evaluation of Operating Data
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient(s) without scram
BOP	balance of plant
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
CAPCO	Central Area Power Coordination (Group)
CAT	Construction Appraisal Team
CEI	Cleveland Electric Illuminating Company
CFR	Code of Federal Regulations
CRD	control rod drive
DCRDR	detailed control room design review
DOE	U.S. Department of Energy
EAL	emergency action level
EBS	emergency broadcast system
ECC	emergency communications center
ECCS	emergency core cooling system
EOF	emergency operations facility
EPI	emergency planning instruction
EPZ	emergency planning zone
ESF	engineered safety feature(s)
ETE	evacuation time estimate
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GDC	General Design Criterion(a)
GE	General Electric Company
HCU	hydraulic control unit
HELB	high-energy line break
HPCS	high-pressure core spray
ICC	inadequate core cooling
ID	inside diameter
IE	Office of Inspection and Enforcement
IFTS	inclined fuel transfer system
INEL	Idaho National Engineering Laboratory
IRM	intermediate range monitoring
ISMG	Instrumentation Setpoint Methodology Group

JIO	justification for interim operation
JPIC	joint public information center
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LRG-II	License Review Group-II
MCPR	minimum critical power ratio
MSIV	main steam isolation valve
MSLB	main steamline break
MSLIV	main steamline isolation valve
NRC	U.S. Nuclear Regulatory Commission
NSRC	Nuclear Safety Review Committee
NSSS	nuclear steam supply system
OBE	operating-basis earthquake
OL	operating license
OSC	operations support center
PASS	postaccident sampling system
PGCC	power generation control complex
PSI	preservice inspection
PVORT	Pump and Valve Operability Review Team
QA	quality assurance
RCIC	reactor core injection cooling
RCPB	reactor coolant pressure boundary
RG	Regulatory Guide
RHR	reactor heat removal
RMT	radiation monitoring team
RPV	reactor pressure vessel
RV	reactor vessel
RWCU	reactor water cleanup
SDV	scram discharge volume
SER	Safety Evaluation Report
SIT	structural integrity test
SOP	standard operating procedure
SPDS	safety parameter display system
SQRT	Seismic Qualification Review Team
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
S/RV	safety/relief valve
SSE	safe shutdown earthquake
SSER	Supplemental Safety Evaluation Report
SV	safety valve
TDI	Transamerica DeLaval, Inc.
TER	Technical Evaluation Report
TMI	Three Mile Island
TSC	technical support center

USGS	U.S. Geological Survey
XLPE	cross-length polyethylene

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Nuclear Regulatory Commission (NRC) Safety Evaluation Report (NUREG-0887) on the application of the Cleveland Electric Illuminating Company (CEI or the applicant) for a license to operate the Perry Nuclear Power Plant (Perry), Units 1 and 2, was issued in May 1982. Supplements to the Safety Evaluation Report (SER) were issued as follows:

Supplement No. 1 in August 1982
Supplement No. 2 in January 1983
Supplement No. 3 in April 1983
Supplement No. 4 in February 1984

The purpose of this Supplemental Safety Evaluation Report (SSER) is to further update the SER by providing the results of the staff's review of information submitted by the applicant by letter addressing some of the issues listed in Sections 1.9, 1.10, and 1.11 of the SER that were not resolved at the time SSER No. 4 was issued. The information provided in these letters must be acceptably documented in amendments to the Perry Final Safety Analysis Report (FSAR) by the applicant before licensing.

Each section or appendix of this SSER is designated and titled so that it corresponds to the section or appendix of the SER that has been affected by the staff's additional evaluation and, except where specifically noted, does not replace the corresponding SER section or appendix. Appendix A is a continuation of the chronology of correspondence between the NRC and the applicant that updates the issues listed in the SER and SSER Nos. 1 through 4. Appendix B is a list of references cited in this supplement.* Appendix C updates the status of unresolved safety issues identified in the SER. Appendix E is a list of the principal contributors to this SSER. Appendix G is a further list of errata to the SER and its prior supplements. Appendices I, J, K, and L are being added to the SER by this supplement. No changes were made to SER Appendices D, F, or H.

In addition to updating the status of unresolved issues, which follows, this supplement

(1) Addresses the staff's acceptance of the Licensing Review Group (LRG)-II proposed solution for the incore instrument tube failure that occurred in the Kuosheng (Taiwan) Nuclear Power Plant, and which is identified as LRG-II Generic Issue 4-MEB (see Section 3.9.2 of this supplement).

*Availability of all material cited is described on the inside front cover of this supplement.

(2) Addresses the staff's acceptance of the Technical Specifications wording changes proposed by the LRG-II relative to Generic Issue 7-CPB, "Rod Withdrawal Transient Analysis" (see Section 15.4.2 of this supplement).

(3) Addresses the staff's favorable findings regarding General Electric (GE) instrumentation setpoints and setpoint methodology program being commissioned by GE Owners Group utilities, including CEI (see Section 7.2.2.8 of this supplement).

(4) Addresses the staff's evaluation findings relative to steam erosion effects on line breaks and valve leakage rates (see Sections 3.6.1 and 10.3.4 of this supplement).

(5) Rescinds the staff's acceptance of the applicant's diesel generator testing program (documented in SSER No. 4) in view of the ongoing staff review addressing the reliability of Transamerica DeLaval diesel generators, used as emergency standby diesel generators in the Perry plant (see Section 8.3.1 of this supplement).

(6) Further revises SER Table 1.1 to add two new generic issues proposed in LRG-II Position Paper VIII, submitted by LRG-II letter dated June 29, 1984, and identifies where in the SER Generic Issues 4-MEB, 3-CSB, and 7-CPB are discussed (see Section 1.12 and Table 1.1 of this supplement).

(7) Addresses the staff's favorable evaluation findings on questions raised during the NRC Construction Appraisal Team's inspection relative to the containment drywell wall structural and bypass leakage integrity due to the installation of approximately 6000-8000 concrete expansion anchor bolts (see Section 3.8.3 of this supplement).

Copies of this SSER are available for public inspection in the NRC Public Document Room at 1717 H Street N.W., Washington, D.C., and at the Perry Public Library, 3735 Main Street, Perry, Ohio.

The NRC Project Manager is John J. Stefano. Mr. Stefano may be contacted by calling (301) 492-7037 or by writing to the following address:

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

Copies of this SSER are also available for purchase from the sources indicated on the inside front cover of this report.

1.9 Outstanding Issues

In Section 1.9 of the SER, the staff identified 19 outstanding issues that had not been resolved at the time the SER was issued. Issues resolved, added, and/or redefined in SSER Nos. 1 through 4 follow:

- SSER No. 1 - Three outstanding issues were reported as being satisfactorily resolved, and Issue (20) was added.

- SSER No. 2 - Three additional outstanding issues were reported resolved, Issue (3) was changed to Confirmatory Issue (53), and Issue (21) was added.

- SSER No. 3 - One outstanding issue was reported resolved, and Issues (22) and (23) were added.

- SSER No. 4 - Three outstanding issues were reported resolved; two issues were reported as being partially resolved; Issues (10) and (12) were changed to Confirmatory Issues (56) and (55), respectively; and Issue (24) was added.

This supplement discusses those issues that have been resolved since SSER No. 4 was issued in February 1984. The composite status of each issue is indicated below. If the issue is discussed in this supplement, the section where it is discussed is identified. Resolution of the remaining outstanding issues will be addressed in a future SER supplement.

Issue	Status	Section
(1) Turbine missile protection	Resolved and added as License Condition (19) in SSER No. 3	---
(2) Seismic system and subsystem analysis	Resolved in SSER No. 1	---
(3) Reactor internals vibration prototype (BWR/6-238 in.) test program	Changed to Confirmatory Issue (53) in SSER No. 2 - resolved in SSER No. 4	---
(4) Environmental qualification of equipment important to safety: (a) Notification that all electrical equipment is qualified or submittal of justification for interim operation (JIO) per 10 CFR 50.49(i) for all unqualified equipment (b) Certification that all mechanical equipment is qualified and submittal of three qualification files for staff review, or provide JIO	Seismic/dynamic qualification of equipment changed to License Conditions (27) and (28); environmental qualification redefined as indicated under "Issues" column and awaiting information from applicant	3.10 and 3.11
(5) Inservice testing of pumps and valves	Resolved and changed to License Condition (26)	3.9.6

Issue	Status	Section
(6) Transient and accident analysis for ECCS, overpressure and operating MCPR	Resolved in SSER No. 1	---
(7) Control room design	Update of review status	18
(8) Mark III containment system (Humphrey issues)	Awaiting additional information from applicant - partial information received under staff review	---
(9) Pool dynamic loads	Partially resolved in SSER No. 4 - awaiting LOCA-related loads information from applicant	---
(10) Containment purge	Changed to Confirmatory Issue (56) and License Condition (24) in SSER No. 4	---
(11) Periodic testing of ADS actuation systems during plant operation	Resolved in SSER No. 2	---
(12) Manual initiation/termination of ESF systems	Changed to Confirmatory Issue (55) in SSER No. 4 - resolved	7.3.2.7
(13) IE Bulletin 79-27	Resolved in SSER No. 4	---
(14) Control system failures	Resolved	7.7.2.1
(15) Fire protection - safe shutdown	Resolved in SSER No. 2 - detailed basis for resolution addressed in SSER No. 3	---
(16) Fire protection - PGCC system (CO_2 vs Halon) fire suppressant in control room	Resolved in SSER No. 2 - detailed basis for resolution addressed in SSER No. 3	---
(17) HPCS skid piping	Resolved in SSER No. 1	---
(18) Interim shift staffing for two-unit operation	Deferred - applies to Unit 2 operation only	---
(19) Emergency plans (onsite)	Changed to Confirmatory Issue (61)	13.3

Issue	Status	Section
(20) Standby liquid control system design	Added in SSER No. 1 - resolved in SSER No. 3	---
(21) Reanalysis of transients and accidents: development of emergency operating procedures per TMI Action Plan Item I.C.1	Added in SSER No. 2 - under staff review	---
(22) Deviations to 10 CFR 50, Appendix R, Section III.F - fire detection requirements	Added in SSER No. 3 - resolved in SSER No. 4	---
(23) FSAR Table 3.2-1 safety-related items list	Added in SSER No. 3 - resolved in SSER No. 4	---
(24) TDI diesel generator reliability	Awaiting information from applicant	---

1.10 Confirmatory Issues

In Section 1.10 of the SER, the staff identified 49 confirmatory issues that were not fully resolved when the SER was issued. Issues resolved, added, and/or redefined in SSER Nos. 1 through 4 follow:

- SSER No. 1 - Five issues were reported resolved, and Issue (50) was added. (Issue (50) was initially cited as License Condition (8) in SER Section 1.11.)
- SSER No. 2 - Twenty-two issues were reported resolved, Issue (6) was deleted, and Issues (51), (52), and (53) were added.
- SSER No. 3 - Six issues were reported resolved.
- SSER No. 4 - Nine issues were reported resolved, Issue (35) was reopened, and Issues (54), (55), and (56) were added. Additionally, the staff's acceptable findings relative to the containment annulus concrete design modification and flaws detected in the steel shell weld radiographs were reported (Issue (3)).

This supplement adds Confirmatory Issues (57), (58), (59), (60), and (61) and discusses those issues that have been resolved since SSER No. 4 was issued. If the issue is discussed in this supplement, the section where it is discussed is identified. Resolution of the remaining confirmatory issues will be addressed in a future SER supplement.

Issue	Status	Section
(1) Piping final stress analysis	Awaiting information from applicant	---

Issue	Status	Section
(2) Containment buckling analysis	Resolved in SSER No. 1	---
(3) Containment ultimate capacity analysis	Resolved in SSER No. 1; staff's acceptance of the concrete "annulus fix" design modification, and flaws detected in steel shell weld radiographs addressed and resolved in SSER No. 4.	---
(4) Emergency service water tunnel structure analysis	Resolved in SSER No. 1	---
(5) Vibration monitoring program for BOP systems	Resolved in SSER No. 1	---
(6) MARK III containment hydrodynamic loads	Deleted in SSER No. 2	---
(7) Testing safety-relief valves per TMI Action Plan Item II.D.1	Under staff review	---
(8) IE Bulletin 79-02	Resolved	3.9.3.3
(9) Dual function pipe whip/ support restraints	Resolved in SSER No. 2	---
(10) Hydrodynamic effect on CRD/HCU	Resolved in SSER No. 4	---
(11) Fuel mechanical fracturing	Resolved	4.2.1.2(8) and 4.2.3.2(8)
(12) Fuel assembly damage from external sources	Resolved	4.2.1.3(4) and 4.2.3.3(4)
(13) Fuel rod bowing	Resolved in SSER No. 3	---
(14) Overheating of gadolinia fuel pellets	Resolved in SSER No. 4	---
(15) Preservice/inservice inspection programs	PSI program partially resolved - awaiting additional information from applicant	5.2.5 and 6.6.3

Issue	Status	Section
(16) Material surveillance program - RV beltline materials	Resolved in SSER No. 2	---
(17) Fracture toughness RCPB materials	Resolved in SSER No. 2 - to be confirmed during Technical Specification review	---
(18) HPCS and RCIC initiation per TMI Action Plan Item II.K.3.13	Site confirmatory audit required to resolve	---
(19) Isolation of HPCS and RCIC per TMI Action Plan Item II.K.3.15	Site confirmatory audit required to resolve	---
(20) Subcompartment pressure analysis	Resolved in SSER No. 3	---
(21) Suppression pool temperature limits	Resolved in SSER No. 4	---
(22) Secondary containment penetration leakage	Resolved in SSER No. 2	---
(23) Containment isolation dependability per TMI Action Plan Item II.E.4.2(f)	Resolved in SSER No. 2 - reopened in this supplement - under staff review	3.9.3.2.1
(24) Type C test of all ECCS injection valves	Resolved in SSER No. 2	---
(25) ADS logic modification per TMI Action Plan Item II.K.3.18	Resolved in SSER No. 4	---
(26) ATWS recirculation pump trip	Resolved in SSER No. 4	---
(27) Modified SDV level monitoring system	Resolved in SSER No. 2	---
(28) HPCS initiation circulatory final design	Resolved in SSER No. 2 - site confirmatory audit required before fuel load	---
(29) Remote shutdown panel nonsafety-grade readouts	Resolved in SSER No. 2	---
(30) RCIC testing procedures	Resolved in SSER No. 2	---

Issue	Status	Section
(31) Calibration of RV/SV pressure switches	Resolved in SSER No. 2 - site confirmatory audit required before fuel load	---
(32) Accident monitoring per TMI Action Plan Items II.F.1.4, II.F.1.5, and II.F.1.6	Resolved in SSER No. 2	---
(33) Failures in vessel level sensing lines common to control and reactor protection systems	Resolved in SSER No. 2	---
(34) Final valve design setpoint and analysis	Resolved in SSER No. 2	---
(35) Physical separation of redundant electrical systems	Resolved in SSER No. 2 - reopened in SSER No. 4 on basis of staff onsite audit - awaiting information from applicant	---
(36) Documentation or test of 3-hour-fire resistance of gypsum board walls	Resolved in SSER No. 4	---
(37) Light and communication fire protection features	Resolved in SSER No. 3	---
(38) Revision of fire protection standpipe and hose locations	Resolved in SSER No. 3	---
(39) Portable fire extinguisher locations	Resolved in SSER No. 3	---
(40) Watertight curbs in switchgear/diesel generator rooms	Resolved in SSER No. 3	---
(41) Design for noble gas effluent monitors per TMI Action Plan Item II.F.1.1	Resolved in SSER No. 4	---
(42) Design for sampling and analysis of plant effluents per TMI Action Plan Item II.F.1.2	Resolved in SSER No. 4	---
(43) Leakage surveillance preventive maintenance program per TMI Action Plan Item III.D.1.1	Changed to License Condition (16) in SSER No. 1	---

Issue	Status	Section
(44) Radiation/shielding design of IFTS tube	Resolved in SSER No.2	---
(45) Location of plant area radiation monitoring per TMI Action Plan Item II.F.1.3	Resolved in SSER No. 2	---
(46) Training program per TMI Action Plan Item II.B.4	Resolved in SSER No. 2	---
(47) Nuclear section training program	Resolved in SSER No. 2	---
(48) Shift supervisor training per TMI Action Plan Item I.C.3	Resolved in SSER No. 2	---
(49) Verify implementation of equipment control measures in radiation areas per TMI Action Plan Item I.C.6	Resolved in SSER No. 2	---
(50) No load, light load, and test loading of the diesel generators	Resolved in SSER No. 2	---
(51) NSSS vendor review of low-power ascension and emergency operating procedures per TMI Action Plan Item I.C.7	Site confirmatory audit required to resolve	---
(52) Pilot monitoring of selected emergency operating procedures per TMI Action Plan Item I.C.8	Analogous to Outstanding Issue (21) - being deleted in this supplement	---
(53) Reactor internals vibration prototype (BWR/6-238 in.) test program	Made a confirmatory issue in SSER No. 2 - resolved in SSER No. 4	---
(54) Preoperational and periodic	Awaiting information on testing plans for the two subsystems in the permanent dewatering system	2.4.9
(55) Labeling LPCI/LPCS injection valve switches to warn operator that inadvertent operation could cause overpressurization	Added in SSER No. 4 - resolved	7.3.2.7

Issue	Status	Section
(56) Containment purge system - details of program to be used to determine purge criteria for life of plant after first refueling cycle, to be submitted 6 months prior to initial fuel load	Added in SSER No. 4 - awaiting purge criteria program details	---
(57) SPDS design requirements per NRC Generic Letter 82-33	Added in this SSER - awaiting information from applicant	---
(58) Confirmation that the lowest temperature will be experienced by the limiting materials of the reactor containment pressure boundary under conditions cited in General Design Criterion 51	Added in this SSER - awaiting information from applicant	6.2.8
(59) Documentation of clarifications to changes made in Chapter 17 of FSAR Amendments 13, 14, and 15 in a future FSAR amendment	Added in this SSER - awaiting FSAR amendment to close	17
(60) Use of meteorology as a part of plant emergency response capability	Added in this SSER - under staff review	2.3.3
(61) Emergency plans (onsite)	Added in this SSER	13.3

1.11 License Conditions

In Section 1.11 of the SER, the staff identified 15 license conditions. These included several issues that must be resolved by the applicant as a condition for issuance of an operating license, and other longer term issues (noted by asterisk) that will be cited in the operating license issued, to ensure that NRC requirements are met during plant operation. Issues resolved, added, and/or redefined in SSER Nos. 1 through 4 follow:

- SSER No. 1 - License Condition (8) was deleted and added to the list of issues in Section 1.10 of the SER as Confirmatory Issue (50), and Confirmatory Issue (43) was made License Condition (16).
- SSER No. 2 - License Condition (17) was added (also listed as Confirmatory Issue (25) in SER Section 1.10) as was License Condition (18). The results of the staff's generic evaluation of License Condition (2) were also presented.

- SSER No. 3 - License Condition (19) was added, the partial findings of the staff's review of information received pertaining to License Condition (4) were reported, and License Condition (4) was redefined.

- SSER No. 4 - License Conditions (20) through (25) were added, and License Condition (1) was rephrased to make it consistent with the statement in Section 2.5.5 of the SER text.

This supplement deletes License Condition (4), redefines License Condition (13), deletes License Condition (18), and adds License Conditions (26, (27), (28), and (29). The updated and current list of the pre- and post-licensing conditions, with references to appropriate SER/SSER sections, is presented below.

(1) Final design of a permanent slope protection system, described in Section 2.4.5.5.3 of the FSAR, will be initiated if the toe or crest of the 3H:1V bluff encroaches closer than 250 ft or 115 ft, respectively, to the emergency service water pumphouse (2.5.5).*

(2) Periodic measurement of channel box deflection must be resolved before startup of the second refueling cycle of operation (4.2.3.1)* - the staff, through its generic evaluation of this license condition, discussed in Section 4.2.3.1 of SSER No. 2, concluded that the LRG-II measures and test program adopted by CEI for Perry would preclude excessive channel bowing in the Perry plant, and that the LRG-II measures and test program will appropriately be referenced in the operating licenses issued for Perry Units 1 and 2.

(3) Operation beyond Cycle 1 is not permitted until stability analyses are provided by the applicant for staff approval (4.4.4).*

(4) The applicant shall implement the staff's requirements regarding additional instrumentation for detection of inadequate core cooling per TMI Action Plan Item II.F.2 based on the staff's review of GE Reports SLI-8211 and SLI-8218 submitted by the BWR Owners Group (BWROG), and the applicant's plant-specific evaluation report addressing the recommendations contained in those GE reports prior to fuel load (4.4.7.1)* - issue is resolved and this condition is accordingly deleted in this supplement (see Section 4.4.7.1 of this SSER).

(5) Hydrogen control for degraded core accidents per TMI Action Plan Item II.B.8 subject to completion of the staff's generic evaluation (6.2.7) - design information is required before fuel load of Unit 1, predicated on the new hydrogen rule, and is awaited from applicant.

(6) IE Bulletin 80-06, engineered safety features reset control (7.3.2.5) - information is required before fuel load of Unit 1 and is awaited from applicant.

(7) Postaccident sampling system per TMI Action Plan Item II.B.3 (9.3.2) - this item was resolved to the staff's satisfaction in SSER No. 4.

(8) No load, light load, and test loading of diesel generators (9.6.3.2) - changed to Confirmatory Issue (50) in SSER No. 1; resolved in SSER No. 2.

(9) Test data to demonstrate that the HPCS diesel generator will not experience undue wear at low room temperatures are to be submitted by the applicant 24 months after fuel load (9.6.4).*

(10) Each operating shift shall be assigned a person with commercial BWR startup/operating experience for a period of 1 year from fuel load, or the attainment of nominal 100% power, whichever occurs later (13.1.2.3).*

(11) Test and maintenance procedures associated with engineered safety features per TMI Action Plan Item II.K.1.5 (13.5.2.3) - information is required before fuel load of Unit 1 and is awaited from applicant.

(12) Procedures for removing safety-related systems from service per TMI Action Plan Item II.K.1.10 (13.5.2.4) - information is awaited from applicant.

(13) The applicant shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans, including amendments made pursuant to the authority of 10 CFR 50.54(p). The approved plans, which contain safeguards information as described in 10 CFR 73.21, are collectively entitled Cleveland Electric Illuminating Company, "Perry Nuclear Power Plant Units 1 and 2 Security Plan," initial submittal undated (transmittal letter dated January 28, 1981), second submittal undated (transmittal letter dated February 25, 1982), Revision 3 dated March 1, 1982 (transmittal letter dated March 22, 1982); "Safeguards Contingency Plan" (Chapter 8 of the Security Plan), initial submittal undated (transmittal letter dated January 28, 1981), Revision 1 dated July 31, 1981 (transmittal letter dated July 31, 1981), Revision 2 dated November 30, 1981 (transmittal letter dated November 25, 1981); "Perry Nuclear Power Plant Units 1 and 2 Security Force Training and Qualification Plan," initial submittal undated (transmittal letter dated January 28, 1981), Revision 1 dated July 31, 1981 (transmittal letter dated July 31, 1981), Revision 2 dated January 8, 1982 (transmittal letter dated January 8, 1982) (13.6).*

(14) Initial test program per TMI Action Plan Item I.G.1 (14).*

(15) Prohibition of extended cycle operation with partial feedwater heating (15.1).*

(16) Leakage surveillance and preventive maintenance program per TMI Action Plan Item III.D.1.1 (11.5) - changed to License Condition (16) in SSER No. 1, formerly Confirmatory Issue (43) in Section 1.10 of the SER. Information is awaited from applicant.

(17) ADS logic modification per TMI Action Plan Item II.K.3 18 (6.3.1.3)* - installation of the staff-approved modification in Unit 1 is required before startup after the first refueling outage; installation in Unit 2 must be completed before initial criticality (the approved modification adopted by CEI for Perry is discussed in Section 6.3.1.3 of SSER No. 4).

(18) Compliance with the guidelines of NUREG-0612 relative to overhead heavy-load-handling system (9.1.5)* - as stated in Section 9.1.5 of SSER No. 2,

before startup following the first refueling outage, the applicant shall comply with the guidelines of Section 5.1.1 of NUREG-0612 (Phase I - the 6-month response to NRC generic letter dated December 22, 1980). Before startup following the second refueling outage, the applicant shall have made commitments acceptable to the staff regarding the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II - 9-month response to generic letter dated December 22, 1980). This condition is being deleted in this supplement (see Section 9.1.5 of this supplement).

(19) Within 3 years of obtaining an operating license, the applicant shall submit for staff approval a turbine system maintenance program based on the turbine manufacturer's calculations of missile generation probabilities. Until the turbine system maintenance program is approved, the applicant shall volumetrically inspect all low-pressure rotors at the second refueling outage and every alternate outage thereafter, and conduct turbine steam valve maintenance (following initiation of power) in accordance with the staff's recommendations as stated in Section 3.5.1.3.1.5 of SSER No. 3 (3.5.1.3.3).*

(20) The applicant is required to have a signed contract with the U.S. Department of Energy (DOE) for nuclear waste disposal services, or the Secretary of Energy must confirm in writing that the applicant is actively and in good faith negotiating with DOE for a contract per the provisions of Section 302(b) of the Nuclear Waste Policy Act of 1982 (P.L. 97-425), before an operating license can be issued for Perry Units 1 and 2 (1.13).

(21) The applicant shall fully implement and maintain all provisions of the fire protection program as described in the FSAR and approved by the staff in the SER and its SSERs as applicable (9.5)* - staff onsite review to be scheduled before fuel load of Unit 1.

(22) The applicant must document and install before fuel load: (a) design features that enable a diesel generator unit in the "test mode" to automatically return to the "emergency standby mode" when a safety-injection signal occurs; and (b) provide an alarm in the dc battery circuit to alert the plant operator(s) of a fuse-open or breaker-open condition in the battery circuits (8.4.4) - information is awaited from applicant.

(23) The applicant shall submit a final report, summarizing the results of the prototype reactor internals test program vibration analyses, measurements, and inspection programs, within 120 days of completion of vibration testing per Regulatory Guide (RG) 1.20 (3.9.2.3).*

(24) The applicant shall provide for staff approval the purge criteria to be used for the remainder of the plant life, based on the results of the programs identified in Section 6.2.4 of SSER No. 4, before startup after the first refueling outage (6.2.4).*

(25) The applicant shall perform the staff-approved Inservice Inspection Program for Class 1, 2, and 3 components before the first refueling outage (6.6.3).*

(26) Pursuant to 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(4)(i), the relief identified in the Pump and Valve Inservice Testing Program for Perry Unit 1 (dated June 15, 1983) is granted during the initial 120-month inspection interval during which period the staff completes its review (3.9.6).*

(27) Resolution of the electrical/mechanical equipment seismic and dynamic qualification confirmatory issues identified in Section 3.10.1 of SSER No. 5 before Perry Unit 1 fuel load (3.10).

(28) Resolution of pump and valve operability assurance program confirmatory issues identified in Section 3.10.2 of SSER No. 5 before Perry Unit 1 fuel load (3.10).

(29) The staff must determine the acceptability of all deviations to Regulatory Guide 1.97, Rev. 1, before Perry Unit 1 fuel load; all plant instrumentation system modification required to comply with Regulatory Guide 1.97, Rev. 2 (including accepted deviations) must be completed before startup following the first refueling outage of Perry Unit 1 (7.5.2.2).*

1.12 Licensing Review Group-II

Table 1.1 of the SER lists issues reviewed or still under review by the staff, jointly submitted by the Licensing Review Group (LRG)-II, a group consisting of the Illinois Power Company, Gulf States Utilities, and the applicant. This table was revised in SSER No. 4 to (1) correct errors found in cross-referencing each issue in the SER text; (2) delete issues no longer being addressed generically by the LRG-II and/or are being addressed by other groups; and (3) add new Issues 5-ASB, 4-MEB, and 3-CSB. Table 1.1 is being further updated in this supplement to add new Issues 4-CSB and 5-CSB and to indicate the sections in this supplement where the staff's evaluation findings relative to Issues 7-CPB, 3-CSB, and 4-MEB are discussed.

Table 1.1 Compilation of LRG-II generic issues (Revision 1)

Issue No.*	Title (TMI Action Plan Item)	SER (SSER) section
1-RSB	Autorestart of HPCS After Manual Termination (II.K.3.21)	7.3.2.1
2-RSB	Design Adequacy of RCIC (II.K.3.13, II.K.3.15, II.K.3.24)	5.4.1, 6.3.1.3
3-RSB	Commitment To Participate in S/RV Surveillance Program	5.2.3
4-RSB	Operator Actions Required 10-20 Minutes Following a LOCA (II.K.3.18)	6.3.3
5-RSB	Control of Post-LOCA Leakage To Protect ECCS and Preserve Suppression Pool Level	6.3.1.3
6-RSB	Applicability of Liquid-Flow-Through-S/RV Test (II.D.1)	5.4.2
7-RSB	Provisions To Preclude Vortex Formation	Appendix C, A-43
8-RSB	Long-Term Air Supply to ADS (II.K.3.28)	6.3.1.3 (2)
9-RSB[a]	Long-Term Operability to ECCS Pumps - Post LOCA	6.3.1.3(4)
10-RSB	LOCA Analysis With Subsequent Flow Control Valve Closure	6.3.3
11-RSB	Use of Nonsafety-Grade Equipment in Shaft Seizure Event	15.2
12-RSB	Classification of Transients: Turbine Trip w/o Bypass and Generator Load Rejection w/o Bypass	15.1
13-RSB	ECCS Valve High Pressure Interlock	6.3.1.3
1-CPB	Ballooning and Rupture of Cladding	4.2.3.3
2-CPB	Seismic and LOCA Loads on Fuel	4.2.3.2, 4.2.3.3
3-CPB	Channel Box Deflection	4.2.3.1(2)
4-CPB	High Burnup Fission Gas Release	4.2.3.3
5-CPB	Cladding Water-Side Corrosion	4.2.3.1

*See footnoted material at end of Table 1.1.

Table 1.1 Revision 1 (Continued)

Issue No.	Title (TMI Action Plan Item)	SER (SSER) section
6-CPB[b]	Inadequate Core Cooling (II.F.2)	4.4.7.2(4)
7-CPB	Rod Withdrawal Transient Analysis	15.4.2(5)
8-CPB	Fuel Analyses for Mislocated or Misoriented Bundles	15.4.3
9-CPB	Discrepancy in Void Coefficient Calculation	4.3.2.2
10-CPB	Rod Worth: Bounding Accident Analysis	15.4.4
11-CPB	Hydrodynamic Stability Analysis (license condition)	4.4.4
1-CSB	Pool Dynamic Loads - swell velocities, etc.	6.2.1.8(4)
2-CSB[b]	Hydrogen Generation and Control	6.2.7
3-CSB[c,d]	Periodic Low Pressure Leakage Testing of the Drywell	6.2.1.7
4-CSB[a,c]	Containment Purge Operational Data Gathering and Evaluation Program	---
5-CSB[a,c]	Containment Access Management Program	---
1-AEB	MSLIV Bypass Leakage Rate	15.3.1
1-ASB	BWR Scram Discharge Volume	4.6
2-ASB[d]	Safe Shutdown for Fires - Sections III.G and III.L of 10 CFR 50, Appendix R	9.5.1
3-ASB	Protection of Equipment in Main Steam Pipe Tunnel - BTP ASB 3-1, revised 7-1-81	3.6.1
4-ASB	Design Adequacy of RCIC Space Cooling	9.4.5.3
5-ASB[c]	CRD System Vessel Inventory Makeup Rate	4.6(3)
1-RAB	Exposure Resulting From Actuation of S/RVs	12.4
2-RAB	Routine Exposures Inside Containment	12.2.1
3-RAB	Controlling Radioactivity During Steam Dryer and Steam Separator Refueling Transfer	12.2.1
4-RAB[d]	Shielding of Spent Fuel Transfer Tube and Canal	12.3.2

Table 1.1 Revision 1 (Continued)

Issue No.	Title (TMI Action Plan Item)	SER (SSER) section
1-ICSB	Failure in Vessel Level Sensing Lines Common to Control and Protectio . Systems	7.7.2.3(2)
2-ICSB	Interlocks Isolating High and Low Pressure Systems	7.6.2.1
3-ICSB	Potential for Both Low-Low Setpoint Valves To Open	7.6.2.2
4-ICSB	Loss of Safety Function After Reset	7.3.2.5
5-ICSB[d]	Control Systems Failure	7.7.2.1(4)
6-ICSB[d]	Procedures Following Bus Failure	7.5.2.4(4)
7-ICSB[d]	Harsh Environment for Electric Equipment Following High-Energy Line Breaks	7.7.2.1(4)
1-PSB	Reliability of Diesel Generators	9.6.3
1-GIB	Interim Licensing Bases Pending Resolution	Appendix C
1-HFS[e]	Special Low-Power Testing Program (I.G.1)	14(1)
2-HFS	ATWS Emergency Operating Procedure and GE Reactivity Control Guidelines (I.C.1)	13.5.2.2
3-HFS[d]	Common Reference for Reactor Vessel Level Measurement (II.K.3.27)	18
1-CHEB[f]	Reactor Coolant Sampling (II.B.3)	9.3.2(4)
2-CHEB[f]	Suppression Pool Sampling (II.B.3)	9.3.2(4)
3-CHEB[f]	Estimation of Fuel Damage From Coolant and Pool Sampling (II.B.3)	9.3.2(4)
1-MEB	Use of SRSS for Mechanical Equipment	3.9.3.1
2-MEB	RPV Internals Vibration Assessment Program	3.9.2.3
3-MEB	OBE Stress Cycles Used for NSSS Mechanical Equipment Design	3.9.1
4-MEB[c]	Kuosheng Incore Instrument Tube Break	3.9.2(5)

Table 1.1 Revision 1 (Continued)

Issue No.	Title (TMI Action Plan Item)	SER (SSER) section
1-MTEB	Inspectability of Welded, Flued Heads	3.6.2
1-SEB[d]	Combination of Loads	---
2-SEB[d]	Fluid/Structure Interaction	---

[a]LRG-II position submitted but not yet addressed in the SER/SSERs.

[b]Being addressed by other groups (e.g., BWR Owners Group, Hydrogen Control Owners Group) in lieu of LRG-II.

[c]Issue added subsequent to publication of the SER (May 1982).

[d]Determined to be nongeneric subsequent to compilation of this list, and has been addressed by CEI on plant-specific basis.

[e]Deleted - no longer a requirement per NRC Generic Letter 83-24 (June 29, 1983); however, justification is required from each LRG-II member in order to relax this test provision of TMI Action Plan Item I.G.1.

[f]Generic Issue 3-CHEB will encompass 1-CHEB and 2-CHEB.

2 SITE CHARACTERISTICS

2.3 Meteorology

2.3.3 Onsite Meteorological Measurements Program

In Section 2.3.3 of the SER, the staff reported its acceptable findings with respect to the applicant's onsite meteorological measurements program, concluding that the system conformed to the guidelines of Regulatory Guide (RG) 1.23 and that the system will provide adequate data as required in 10 CFR 100.10. However, with respect to the use of meteorology as a means for emergency response capability, the staff requested its consultant to review the meteorological aspects provided by the applicant in NUS Corporation Report No. NUS-4336 (April 1983), entitled "Description of Perry Nuclear Power Plant Emergency Offsite Dose Calculations." By letter dated August 22, 1984, the applicant was advised of the staff consultant's conclusions and recommendations which follow:

Conclusions:

(1) It is not apparent how the applicant proposes to handle the measurement of a spatial variable, three-dimensional trajectory on a real-time basis in view of the fact that the only wind data available are the horizontal wind speed and direction at the 10-m and 60-m levels of the onsite meteorological tower. No other wind measurement sites within the emergency planning zone (EPZ), a circle of 10-mi radius, are listed as being available.

(2) Although the extensive use of site-specific algorithms is mentioned as input to the EMERGE Code, none are specifically described or justified.

Recommendations

(1) The NUS Corporation report should be expanded to include a full description and justification of the meteorological measurements that are used.

(2) Supplemental measurement capability should be considered in order to provide real-time input to the various meteorological algorithms used in the EMERGE Code program.

By letter dated January 14, 1985, the applicant provided a response to the above consultant conclusions and recommendations, which are currently being reviewed by the staff. Until this matter is fully resolved, it is being added in this supplement to the list of issues in Section 1.10 of the SER as Confirmatory Issue (60).

2.4 Hydrologic Engineering

2.4.9 Conclusion

In Section 2.4.9 of SSER No. 4, the staff introduced Confirmatory Issue (54) requiring the applicant to provide preoperational and periodic testing plans for the two subsystems in the permanent dewatering system. By letter dated September 21, 1984, the applicant furnished additional information toward resolving this issue. As part of its review of the applicant's submittal, the staff visited the plant site to inspect the underdrain system and test facilities and to discuss test procedures with the applicant and his consultant. This inspection included a descension into one manhole to inspect the gravity drain piping and the dewatering pump. The test facilities (piping, valves, etc.) were all found to be temporary and were not in place. Some of the apparent discrepancies discussed were as follows:

(1) The Functionability Test Plan (Test Plan) does not discuss blocking of the 12-in. porous concrete pipe in the east-west direction during the tests. This pipe should be blocked at manholes 1, 8, 9, and 14.

(2) The Test Plan describes a constant inflow and outflow of 50 gpm and then the stabilization of the system piezometric surface in the east-west direction. This does not allow for the constant groundwater inflow of about 30 gpm into the system, and thus the piezometric surface will never stabilize. The test procedure must be changed so that the inflow is constant, but the outflow will vary until the piezometric surface is stabilized. The measured outflow is to be recorded.

(3) Tables 2.4-9 and 2.4-10 of the FSAR should specify elevations for the values quoted.

It will not be required that the test be performed for high flow (30,000 gpm) capacity of the gravity drain systems; only periodic inspection to ensure lines are not blocked will be required.

The applicant has agreed to revise and correct the test procedures accordingly and to resubmit the Test Plan for staff acceptance before Unit 1 fuel load. The staff's findings will be documented in a future SER supplement. Therefore, Confirmatory Issue (54) continues to remain unresolved.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

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

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

By letter dated July 22, 1983, the applicant was requested to provide information relative to the effects of steam erosion on Perry plant components. This request was prompted by licensee event reports and the NRC Office of Inspection and Enforcement information notices which reported piping and component (main steam isolation valve leakage) failures in operating reactor plants presumed to have been caused by steam erosion phenomena. In letters dated August 31, 1983, and May 15, 1984, the applicant responded to the staff's information request; in particular, the applicant's May 15, 1984, letter transmitted an independent and comprehensive study entitled "Steam Erosion Hazards Analysis," which contained information concluding that the safe shutdown of the Perry plant would be ensured if a steam erosion-related pipe break were to occur.

In a letter dated August 28, 1984, the staff identified the need for additional information predicated on its then ongoing review of the applicant's "Steam Erosion Hazards Analysis" report. The applicant provided this additional information in a letter dated September 18, 1984.

The staff has since completed its review of the information furnished in the above-mentioned applicant correspondence and has concluded that the Perry design acceptably prevents and mitigates the effects of steam erosion, and that the applicant will establish an acceptable inservice inspection program to monitor the piping system for signs of steam erosion-related failures. Details of the staff's evaluation findings, resulting in this conclusion, are presented below relative to plant piping systems outside containment and in Section 10.3.4 of this supplement with respect to steam erosion effects on main steam isolation valve leakage.

According to H. Keller, in his technical paper entitled "Erosion-Corrosion in Wet Steam Turbines," steam erosion of carbon steel occurs in "wet" flowing steam at temperatures below approximately 480°F. The design of BWRs deals with this situation by providing steam separator and dryer assemblies in the upper reactor head region. These assemblies produce high-temperature (535°F-550°F), low-moisture steam for all steam flow conditions. This steam exits the reactor vessel at approximately 550°F and passes through the main steamlines to the turbine at temperatures ranging from 535°F to 540°F. As the steam flows through the turbine, it loses energy and becomes cooler. This cooling of the steam results in condensation, which increases the moisture content of the steam. The steam is extracted (diverted) at six points in the main steam cycle: for regenerative feedwater heating, for driving the reactor feedwater pump turbine, and for generating seal steam for the steam evaporator during normal operation. The steam is also diverted from the main steamline for reheating purposes, offgas preheaters, and steam jet air ejectors.

On the basis of its review of the Perry main steam system, the staff has determined that the main steamlines that supply steam to the main turbine are not exposed to steam erosion because of the high-temperature (535°F-550°F), low-moisture content of this steam. Because this high-temperature, low-moisture steam also passes through the main steam systems branch lines that supply steam for reheat purposes, seal steam generation, and offgas preheaters, these branch lines are likewise not exposed to steam erosion. As such, steam erosion is not expected to occur in these safety-related portions of the main steam system.

The lines that are exposed to conditions under which steam erosion can occur over a period of time are the drains of equipment and steamlines, and the steam extraction lines at the main turbine. In these locations, the steam will have cooled considerably below the initial reactor outlet temperature of 550°F with an accompanying increase in moisture content. Operating plant experience described in the NRC Office of Analysis and Evaluation of Operating Data report entitled "Erosion in Nuclear Power Plants" (June 11, 1984) - which assesses events reported in licensee event reports and Office of Inspection and Enforcement information notices (more specifically IE Information Notices 82-22 and 82-23), identifying line breaks and valve leakages, respectively, caused by steam erosion - reflects this, in that the majority of the steam erosion events occurred in the steam extraction drain lines and feedwater heating systems. These plant areas are in nonsafety portions of the main steam system, and steamline breaks in those lines do not constitute a safety concern to the staff.

The staff also examined the plant areas through which steam pipes pass, which are believed to be susceptible to steam erosion, and found that all of these lines are separated from equipment necessary to safely shut down the plant. Thus, the failure of these lines will have no effect on bringing the plant to a safe shutdown condition.

As additional "defense-in-depth," action has been taken by the applicant to (1) provide an extra margin of the thickness in the piping walls to allow for erosion; (2) establish an inservice inspection program to monitor the effects of steam erosion in extraction steam piping; and (3) analyze the consequences of steam erosion at the Perry plant, the results of which are presented in the applicant's "Steam Erosion Hazards Analysis" report, mentioned above.

(1) Wall Thickness Margin

Good piping practice dictates that extra piping wall thickness be provided to allow for steam erosion. In a letter dated May 15, 1984, the applicant stated that the maximum material loss due to steam erosion effects for the Perry steamlines over the 40-year life of the plant is expected to be well within the erosion allowance of the pipe design for those systems. The applicant's erosion allowances exceed the corresponding minimum wall thickness by 50 to 400%. The engineering basis for the applicant's calculations for erosion of carbon steels is an empirical formulation based on operating experience developed by H. Keller. The staff has reviewed H. Keller's technical paper and concludes that the functional relationships shown between various parameters in the equations affecting steam erosion appear reasonable.

(2) Inservice Inspection

In addition to providing the main steam system piping with extra margin in wall thickness to allow for steam erosion effects, the applicant has committed to initiate an inservice inspection program to monitor the effects of erosion in the extraction steam piping. The extraction steam piping was chosen since it is the locale where the worst steam erosion is expected to occur. Steam erosion is maximized in this locale because high-moisture steam is the cause of steam erosion and the moisture content of the steam is maximized as it flows through the turbine into the extraction steam piping. To perform this inservice inspection program more easily, large plates with telltale valves have been placed in critical locations in the steam extraction piping. The telltale valves will be used to determine if steam erosion remains within the design limits. In addition to the telltale valves, the applicant has committed to periodically determine the actual extraction steam piping wall thickness by using radiographic or ultrasonic techniques.

(3) Effects Of Steam Erosion

Lastly, as part of the original pipe break review as required by Standard Review Plan (SRP) Section 3.6.1 (NUREG-0800), the applicant performed a comprehensive study to assess the effects of piping failures in the piping systems where steam erosion is likely to occur, namely the drains of equipment and steamlines and the extraction lines at the main turbine. The applicant stated that all safety-related piping systems and components in the Perry plant are separated from the areas containing piping systems where steam erosion is likely to occur.

For the lines where the reactor design prevents steam erosion by producing dry high-temperature steam, the applicant provided an analysis that goes beyond the criteria of SRP Section 3.6.1. For these piping systems the applicant demonstrated that wherever a piping failure was postulated and safety-related equipment was affected, there was sufficient equipment unaffected by the break to mitigate the consequences of the break and bring the reactor to the safe shutdown condition. The staff agrees that the study performed by the applicant does show significant capability to bring the reactor to the safe shutdown condition.

Conclusion

From its review of the main steam system of the Perry plant, the staff concludes that the safety-related portions of the system are exposed to high-temperature (535°F-550°F), low-moisture content steam and are not subjected to steam erosion. The lines that the staff finds to be exposed to moist steam for a sufficient period of time to cause erosion are the drains of equipment and steamlines and the steam extraction lines at the main turbine. The staff found that these non-safety-related lines are separated from equipment that is necessary to shut down the plant and therefore concludes that the failure of these lines would have no adverse effect on bringing the plant to a safe shutdown condition. Additional assurance of plant safety is provided by defense-in-depth steps that have been taken by the applicant to ensure safe operation over the 40-year life of the plant through the provision of an extra margin in the steam piping wall thickness, an inservice inspection program that monitors the steam erosion effects in the piping where the steam erosion effects are expected to be maximized, and the analysis that assesses the effects of piping failures where steam erosion is likely to occur.

3.8 Design of Seismic Category I Structures

3.8.3 Concrete and Structural Steel Internal Structure

During the inspection of the Perry plant by the NRC Construction Appraisal Team (CAT) in August-September 1983, a concern was identified by the CAT inspectors relative to the containment drywell wall structural and bypass leaktightness integrity as a result of the installation of concrete expansion anchor bolts (Hilti-Kwik bolts). The CAT concern, documented in the CAT inspection report submitted to the applicant by letter dated November 7, 1983, about the number of anchor bolts installed (6000-8000 bolts) and the potential for throughwall cracks during normal, test, transient, and accident conditions was whether the drywell was capable of meeting structural requirements and the bypass leak limits stated in SER Section 6.2.1.7. In a letter dated May 30, 1984, the applicant was provided several questions pertaining to this concern, which he was requested to address. By letter dated September 19, 1984, the applicant provided responses to the specific questions raised by the staff. Following is the staff's evaluation of those responses:

(1) Installation of Expansion Bolts and Crack Control Around Bolts

The vast majority of expansion bolts installed in the Perry drywell wall fall into three categories:

(a) 5/8-in.-diameter bolts with embedded depth of approximately 4 in.: Minimum spacing of bolts is approximately 3 in. The maximum allowable tension capacity is 2.83 kips, and the shear capacity is 3.13 kips.

(b) 3/4-in.-diameter bolts with embedded depth of approximately 4 in.: Minimum spacing of bolts is approximately 3 in. The maximum allowable tension capacity is 3.38 kips, and the shear capacity is 4.40 kips.

(c) 3/4-in.-diameter bolts with embedded depth of approximately 7 in.: Minimum spacing of bolts is 4 in. The maximum allowable tension capacity is 5.56 kips, and the shear capacity is 4.93 kips.

Each bolt has (leak) sealer tape installed in the annular space between the bolt shank and the oversize hole in the ¼-in. liner plate to provide leaktightness around each bolt. Because the as-built drywell wall is 5 ft thick with an inside "clear" concrete cover over the reinforcing bars of 5½ in., the staff concurs with the applicant's observation that the installed expansion bolts are unlikely to impose any interference with reinforcement. Also, the load limits placed on the majority of expansion bolts have been restricted to be less than approximately 250 lb tension and shear. Both the normal operating and the accident temperature gradients through the drywell wall tend to keep the inside of the drywell wall in compression, thus reducing a potential for crack initiation or propagation.

(2) Design of Drywell Liner

At Perry, the drywell liner is provided as a form for the concrete. However, the drywell liner and its anchorage system were designed to conform with CC-3000 of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Division 2. Installation of concrete expansion bolts will not affect the stress, strain, and displacement limits of liner plates because the bolt loads are transferred directly to the concrete. On the basis of available General Electric concrete cracking studies and plant-specific calculations, the applicant concluded that the concrete drywell wall alone, without the assistance from the liner, can be considered leaktight.

Therefore, the staff agrees with the applicant that the liner is not required to meet the bypass leakage requirement of the FSAR.

(3) Assessment of Potential Throughwall Crack on As-Built Drywell Wall Structure and Bypass Leakage

An assessment performed by the applicant showed that no throughwall concrete cracks have developed in the as-built drywell wall. The applicant observed that only minor surface cracking exists as a result of drying shrinkage of the concrete. The applicant also demonstrated, by calculation, that the drywell wall is not expected to develop throughwall cracks. The calculation is based on the high concrete strength, high reinforcement strength, and the ratio of steel area/concrete area used in the design of the drywell wall structure. Moreover, special reinforcement details (added reinforcement) were provided around all penetrations to control local cracking and strain.

The applicant's calculation also demonstrated that under a small-break accident, no bypass leakage flow path is predicted through the drywell wall. Under a design-basis accident, even with extremely conservative assumptions of peak drywell pressures, maximum safe shutdown earthquake loading, safety/relief valve actuations, and precracks at all construction joints, the maximum estimated bypass leakage flow path is only 0.35 ft^2, which is much less than the allowable limit of 1.68 ft^2.

On the basis of these facts and the knowledge that very small loads will be imposed on the expansion bolts, the staff concurs with the applicant's assessment that it is unlikely that through-concrete cracks will develop in the drywell wall structure.

In addition, the applicant has committed to perform the structural integrity test (SIT) at the design pressure of 30 psi to demonstrate that the drywell wall structure fully complies with the FSAR commitments and to verify that the as-built drywell wall has been designed and constructed to maintain its leaktight integrity.

Conclusion

The staff has reviewed the information and the assessment provided by the applicant and finds that, since the load imposed on the installed expansion bolts will be very small, it is unlikely that the bolt loads will contribute to initiation and propagation of concrete cracks. Analytical calculations based on conservative design also indicated that there will be no throughwall cracking and, therefore, no significant bypass leakage flow paths through the drywell wall. Considering these facts and the applicant's commitment to perform an SIT to demonstrate, among other things, leaktight integrity requirements, the staff concludes that installation of the expansion bolts will not impair integrity of the drywell wall structure.

3.9 Mechanical Systems and Components

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

In a letter dated May 7, 1982 (C. A. Cameron to W. R. Mills), the General Electric Company (GE) informed the NRC that breakage of an incore instrument tube had occurred during a normal shutdown of the Kuosheng 1 Nuclear Power Plant (Kuosheng), located in Taiwan, as a result of operation of the reactor heat removal/low pressure coolant injection (RHR/LPCI) system in an abnormal mode for an extended period of time. The failure was caused by the LPCI system flow injection into the core, which resulted in fatigue failure of the incore instrument tube and subsequent leakage (appoximately 1 gpm) from the reactor vessel. GE stated that such an occurrence could only happen in BWR/6 designed plants where the RHR/LPCI piping is connected to the core shroud below the top guide plate, causing tube vibration; earlier BWR designs have the RHR/LPCI piping connected to the core shroud above the top guide plate so that LPCI flow does not directly impinge on the incore instrument tube.

By letter dated May 18, 1982 (R. Artigas to R. Tedesco), GE provided the NRC staff with their plan of action to eliminate this problem in all BWR/6 plants. The Licensing Review Group (LRG)-II BWR/6 plant owners, including the Perry applicant, elected to work directly with GE to resolve the problem on a generic basis and accordingly added this issue to the list of other LRG-II generic issues (see Table 1.1 of the SER and SSER No. 4) as Generic Issue 4-MEB, "Kuosheng Instrument Tube Break."

By letter dated September 3, 1983 (D. L. Holtzsher to T. A. Novak), the LRG-II provided for NRC review and acceptance, their generic solution for Generic Issue 4-MEB. The LRG-II proposed the following design modifications to the BWR/6 reactor internals to prevent the incore instrument tubes in their plants from experiencing the flow-induced vibration and fatigue problem that occurred at Kuosheng:

(1) Install a flow deflector on each of the three LPCI inlets to prevent direct cross-flow impingement on the core and incore instrument tube - The flow deflector would be a rectangular-shaped plate (approximately 1 ft by 2 ft) with a conical flow splitter to redirect LPCI flow in two horizontal directions, tangential to the core. The deflectors would be fabricated of 316L stainless steel material and attached to the core shroud wall by full-penetration welds at the deflector legs at the four corners of the plate. The processes and procedures to be used for fabrication will be comparable to those used for other reactor internals.

(2) Remove the thermal shields used to protect the LPCI nozzle attachment ring - As originally designed for BWR/6 plants, the thermal shields were installed to protect the LPCI nozzle attachment ring, attaching the LPCI pipe to the core shroud, and served as load-bearing members. A high-flow test of the design modification with the flow deflector, performed by GE, resulted in failure of the thermal shield because of the presence of the deflector. A more substantial strut was added in the design to perform

the load-bearing function provided by the attachment ring. Therefore, GE proposed removal of the thermal shield to the LRG-II. Because removal of the thermal shield would only impact the attachment ring, no longer depended on as a load-bearing structure, its removal would not be critical. The temperature differential across the attachment ring was calculated to be 200°F under normal conditions, and 450°F with the LPCI functioning under accident conditions. Thermal fatigue analysis of the combined LPCI strut (the new load-bearing member) and the attachment ring, with the thermal shield removed, showed that removal of the thermal shield would be acceptable. Subsequent GE tests, which showed that additional stresses on the attachment ring as a result of flow-induced vibration with the thermal shield removed, when combined with thermal stresses, did not have any adverse impact, verified the acceptability of removing the thermal shields. In addition, should the attachment ring fail, the LPCI function would not be impaired, and any leakage resulting therefrom would be into the annulus region around the core and shroud and would not affect total makeup rate to the vessel.

(3) Replace the intermediate range monitoring (IRM) tube nearest each LPCI inlet with a strengthened tube of improved design - This modification would prevent IRM tube damage in the immediate vicinity of the LPCI injection point with the deflector installed and the LPCI operative for extended periods of time.

By letter dated September 9, 1983 (D. L. Holtzscher to J. J. Stefano), the LRG-II supplemented their design modification position (i.e., the one in the September 3, 1983, submittal) by proposing the following operational controls:

(1) Plant operators will be instructed not to operate the RHR system in the LPCI mode with flow to the vessel unless it is required for an accident, emergency, or short-term testing situation as delineated in plant operating procedures.

(2) Should the modified RHR/LPCI system be inadvertently operated with flow to the vessel for an extended period of time, the circumstances will be reported to the NRC in accordance with emergency core cooling system (ECCS) Technical Specification requirements for special reporting of ECCS injection into the reactor coolant system.

The staff has reviewed the LRG-II solution for Generic Issue 4-MEB and finds that the design modification proposed for the RHR/LPCI system is acceptable and that the analyses and tests performed by GE for the LRG-II plants (and a sister non-LRG-II BWR/6 plant, Grand Gulf Nuclear Station) supporting the design modification verify that the modified configuration will have vibration levels below the endurance limit for fatigue, and adequate margins to sustain the combined effects of hydrodynamic, thermal, and seismic loads exist. Installation of the strut near the LPCI inlet has reduced the significance of the attachment ring as a load-bearing member, negating the need for the thermal shield. In addition, the staff finds the procedural controls to be implemented on the operation of the RHR/LPCI system will provide adequate measures to prevent the occurrence of an incident similar to that in Kuosheng in LRG-II plants.

The staff's generic acceptance of the proposed solution for Generic Issue 4-MEB was formally transmitted to the LRG-II in a letter dated December 14, 1983

(B. J. Youngblood to D. L. Holtzscher), advising of the need for each LRG-II member to endorse implementation of the accepted design modification to have this issue considered resolved in their respective plant SERs. In a letter dated May 25, 1984, the applicant endorsed the staff's generic acceptance of the solution for Generic Issue 4-MEB for implementation at Perry. Generic Issue 4-MEB is therefore resolved as an issue for Perry.

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

3.9.3.2 Design and Installation of Pressure Relief Devices

3.9.3.2.1 Pump and Valve Operability Assurance

In Section 3.9.3.2.1 of the SER, the staff reported that TMI Action Plan Item II.E.4.2, as it relates to purge and vent valve operability, had not been completed by the applicant and that information provided by the applicant would be addressed by the staff in an SER supplement. This item was identified as part of Confirmatory Issue (23) in Section 1.10 of the SER, which encompassed other provisions of TMI Action Plant Item II.E.4.2 concerning containment isolation dependability. In Section 6.2.4 of SSER No. 2, the staff reported its evaluation findings relative to the containment isolation design of the plant, concluding that Confirmatory Issue (23) has been resolved. However, it was later determined that the basis for resolution did not address Subitem (f) of TMI Action Plan Item II.E.4.2 as it relates to purge and vent valve operability. This area of containment isolation dependability still needs to be resolved. Consequently, Confirmatory Issue (23) is being reopened in this supplement.

By letter dated December 19, 1984, the applicant provided information for the resolution of this confirmatory issue, that is, purge and vent valve operability documentation. The staff's findings will be reported in a future SER supplement.

3.9.3.3 Component Supports

In Section 3.9.3.3 of the SER, the staff considered the applicant's response to Office of Inspection and Enforcement (IE) Bulletin 79-02, concerning pipe support base plate flexibility and its effects on anchor bolt loads, to be a confirmatory issue, pending completion of its review of the applicant's responses to that bulletin dated July 6, 1979, January 11, 1980, and December 19, 1980.

The NRC regional staff conducted an onsite audit of actions taken by the applicant to comply with IE Bulletin 79-02 and found that the applicant has appropriately addressed the elements of IE Bulletin 79-02 in accordance with applicable requirements. A report of the staff's findings and conclusions is included in IE Inspection Reports 50-440/84-20 (DRS) for Unit 1 and 50-441/84-18 (DRS) for Unit 2, dated September 13, 1984, which are summarized as follows:

(1) The applicant performed a detailed review in accordance with the subject criteria of IE Bulletin 79-02 on a representative sample of safety-related (seismic Category I) supports and analyzed the possible effects of base plate flexibility on base plate anchors.

(2) The staff determined that most plates were flexible as defined by IE Bulletin 79-02 criteria. Therefore, the applicant reanalyzed the plates using a method by which the effects of plate flexibility, anchor preload, and shear-tension interaction were considered. The results of the reanalysis generally confirmed the adequacy of the original design.

(3) A representative sample consisting of 10 Perry-specific designs and 96 similar designs were investigated by the staff. The Perry-specific and similar designs considered were anchored with Hilti-Kwik bolts. The analytical investigations indicated that approximately 5% of the Perry designs may have had a factor of safety less than 4.0 when plate flexibility was considered.

(4) The applicant committed to perform analytical work and appropriate redesign to ensure that all pipe support base plates conform to the requirements of IE Bulletin 79-02 and ASME Code, Section III. This was included in the design verification efforts for seismic Category I supports.

(5) All seismic Category I supports are potentially subject to a relatively low number of seismic loading cycles that can be accommodated by the design. Operational loads that could, during the lifetime of the plant, undergo a large number of load cycles are to be identified during startup testing, and modifications to the pipe support system are to be made as required to ensure that such loads are accounted for.

(6) The results of the investigation of the effects of plate flexibility on pipe support base plate anchors indicate that, for most plates anchored to concrete surfaces with Hilti-Kwik bolts, prying forces did not exist. Prying forces were found to be present in approximately 5% of the cases. In those cases the prying was responsible for an average increase in the bolt tension of less than 30%.

(7) All base plates for large-bore (2 1/2 in. and larger) Safety Category I pipes have been or are being reanalyzed. There are approximately 500 base plates in the two units that fit this category. Small-bore (2-in. and smaller diameter) pipe was designed using seismic support spacing criteria. The criteria were developed on the basis of a conservative pipe stress and a multispan model for each pipe size and schedule. The model analysis provides pipe spans and support loads. This approach has been verified by sample computer analyses to be conservative relative to applicable code requirements.

A series of typical support designs was generated and load rated by analytical techniques. The supports were analyzed for structural adequacy for all members, welds, and expansion anchor bolts. In generating the load rating, the most conservative geometrical combination of the maximum distance from the pipe to the structure was used in conjunction with the smallest allowed spacing between expansion anchor bolts. This resulted in the worst load case. The results of this conservative approach indicate that about 15% of the supports on any of the small-bore piping runs could fail and the piping stresses would remain within code allowable limits. Therefore, detailed analyses and inspection of these expansion anchor bolts were considered unnecessary.

(8) The applicant has appropriately addressed the elements of IE Bulletin 79-02 in accordance with the requirements.

In view of the above related inspection findings and conclusions, Confirmatory Issue (8), listed in Section 1.10 of the SER, is considered to be resolved.

3.9.6 Inservice Testing of Pumps and Valves

To ensure that all ASME Code, Class 1, 2, and 3 safety-related pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant, a test program will be conducted which includes baseline preservice testing and periodic inservice testing. The program provides for both functional testing of the components in the operating state and for visual inspection for leaks and other signs of distress. The inservice testing program for the pumps and valves will meet the requirements of 10 CFR 50.55a(g), including the 1980 Edition of the ASME Code, Section XI, through the Winter 1980 Addenda.

In Section 3.9.6 of the SER, the staff identified the absence of the applicant's program for inservice testing of pumps and valves as an outstanding issue. By letter dated June 15, 1983, the applicant submitted the required program, requesting relief from the above-mentioned ASME Code requirements pursuant to 10 CFR 50.55a(g)(1), for certain pump and valve tests.

At this time, the staff has not completed its detailed review of the applicant's testing program; however, the staff has evaluated the applicant's request for relief and finds that it is impractical, within the limitations of design, geometry, and accessibility, for the applicant to meet these ASME Code, Section XI, testing requirements for which relief has been requested. Imposition of full compliance with these requirements would, in the staff's view, result in hardships or unusual difficulties without a compensating increase in the level of quality of safety. Therefore, pursuant to 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(g)(4)(i), the relief identified in the Pump and Valve Inservice Testing Program for Perry Unit 1, dated June 15, 1983, is granted during the initial 120-month inspection interval during which period the staff completes its review. Consequently, that portion of SER Outstanding Issue (5), pertaining to the inservice testing program for pumps and valves, is being made License Condition (26) by this supplement.

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

3.10.1 Seismic and Dynamic Qualification

As part of the review of FSAR Sections 3.9.2 and 3.10, an evaluation is made of the applicant's program for seismic and dynamic qualification of safety-related electrical and mechanical equipment. The evaluation consists of (1) a determination of the acceptability of the procedures used, standards followed, and the completeness of the program in general and (2) an audit of selected equipment to develop a basis for the judgment of the completeness and adequacy of the seismic and dynamic qualification program.

Guidance for the evaluation is provided by the Standard Review Plan (NUREG-0800), Section 3.10, and its ancillary documents, RGs 1.100, 1.61, 1.89, and 1.92, NUREG-0484, and Institute of Electrical and Electronic Engineers (IEEE) Standards 344-1975 and 323-1974. These documents define acceptable methodologies for the seismic qualification of equipment. Conformance with these criteria is required to satisfy the applicable portions of General Design Criteria (GDC) 1, 2, 4, 14, and 30 of Appendix A to 10 CFR 50, as well as Appendix B to 10 CFR 50 and Appendix A to 10 CFR 100. Evaluation of the program is performed by a Seismic Qualification Review Team (SQRT), which consists of engineers from the NRC and the Idaho National Engineering Laboratory (INEL) (EG&G Idaho). The INEL report, which provides further details of the audit findings, is being added to the SER as Appendix I by this supplement.

The SQRT has reviewed the equipment dynamic qualification information contained in FSAR Sections 3.9.2 and 3.10 and made a visit to the site from August 14 through August 17, 1984. The purpose was to determine the extent to which the qualification of equipment, as installed at Perry Unit 1, meets the criteria described above. A representative sample of safety-related electrical and mechanical equipment, as well as instrumentation, included in both nuclear steam supply system (NSSS) and balance-of-plant (BOP) scopes, was selected for the audit. Table 3.1 identifies the equipment audited. The site visit consisted of field observations of the actual, final equipment configuration and its installation. This was followed by a review of the corresponding design specifications and test and/or analysis documents which the applicant maintains in his central files. Observing the field installation of the equipment is necessary to verify and validate equipment modeling employed in the qualification program. In addition to the document reviews and equipment inspections, the applicant presented details of the qualification and inservice inspection programs.

On the basis of the observation of the field installation, review of the qualification documents, and responses provided by the applicant to SQRT's questions during the audit, the staff finds that the applicant's seismic and dynamic qualification program is well defined and adequately implemented. Upon closure of the issues identified in Table 3.1, and provided that the conditions delineated in the following section are met, the seismic and dynamic qualification of safety-related equipment at Perry Unit 1 meets the applicable portions of GDC 1, 2, 4, 14, and 30, Appendix B to 10 CFR 50, and Appendix A to 10 CFR 100.

The granting of an operating license is dependent on the resolution of the following issues, which is being added to Section 1.11 of the SER by this supplement as License Condition (27):

(1) The applicant must supply confirmation that all safety-related equipment has been fully qualified. This requirement may be waived for a limited number of items, provided that Justifications for Interim Operation have been submitted and approved for all unqualified safety-related equipment before granting of the license.

(2) New hydrodynamic loads (related to the loss-of-coolant accident (LOCA)) have been calculated and approved by the NRC. The impact of the new loads on the qualification of equipment must be assessed. A schedule for the assessment and confirmation that the affected equipment has been qualified under the new loads is needed. (This is related to Outstanding Issue (9), listed in Section 1.9 of the SER and this supplement.)

(3) The question of whether the functioning of the rod multiplexer cabinet is safety related or not must be resolved. If safety related, the qualification must be upgraded to the required level.

3.10.2 Operability Qualification of Pumps and Valves

The NRC staff performs a two-step review of each applicant's pump and valve operability assurance program to determine whether the applicant's program is adequate to ensure that pumps and valves important to safety will operate when required during the life of the plant under normal and accident conditions. The first step is a review of FSAR Section 3.9.3.2. However, the information provided in this FSAR section is general in nature and by itself is not detailed enough to adequately determine the scope of the overall equipment qualification program as it pertains to pump and valve operability. Therefore, in addition to an FSAR review, a Pump and Valve Operability Review Team (PVORT) consisting of engineers from the NRC and the Idaho National Engineering Laboratory (EG&G Idaho) conducts an audit of a representative sample of installed pump and valve assemblies and supporting documentation at the plant site.

This onsite audit is a necessary second step that permits the PVORT to assess the overall program, as implemented, and thereby determine whether the program conforms to the current licensing criteria presented in Section 3.10 of the Standard Review Plan (SRP). Conformance with SRP Section 3.10 criteria is required to satisfy the applicable portions of GDC 1, 2, 4, 14, and 30 of Appendix A to 10 CFR 50 as well as Appendix B to 10 CFR 50.

The PVORT has reviewed the pump and valve operability assurance information contained in Section 3.9.3.2 of the Perry FSAR and conducted an onsite audit from August 14 through August 17, 1984, to determine the extent to which the pumps and valves important to safety, as installed at Perry Unit 1, meet the criteria described above. The onsite audit consisted of field observations of the final equipment configuration and installation of a representative sample of both NSSS and BOP pumps and valves. The field observations were followed by a review of the design and purchase specifications, test and/or analysis documents, and other documents related to equipment operability which the applicant maintains in his central files. Table 3.2 identifies the equipment that was audited. In addition, the applicant presented details of the overall equipment qualification program as well as those programs necessary to ensure that equipment qualification issues and concerns would continue to be addressed.

The staff concludes that the equipment qualification personnel for Perry are dealing with the equipment qualification issue in a very positive manner. On the basis of the confirmation of the successful completion of the qualification program, as it relates to pump and valve operability, the staff concludes that the applicant has provided an acceptable means of meeting the applicable portions of GDC 1, 2, 4, 14, and 30 as well as Appendix B to 10 CFR 50.

On the basis of the results of the component walkdown and review of the qualification document packages performed at Perry on August 14-17, 1984, as well as the explanations provided by the applicant throughout the audit, the staff concludes that an appropriate pump and valve operability assurance program has been defined and implemented. The continuous implementation of this overall program should provide adequate assurance that the equipment items under this program will perform their safety-related functions as required over the life of the plant.

Table 3.2 summarizes the qualification status of each equipment item selected for the pump and valve operability assurance program review. There are no specific concerns regarding the equipment reviewed during the audit. However, there are generic issues that could affect the operability assurance of all pumps and valves. These generic issues (identified below) are confirmatory in nature and must be completed by the applicant before fuel load. The qualification status of each equipment item is considered to be closed pending confirmation by the applicant that these conditions are met.

The staff requires that all safety-related equipment shall be qualified and approved by the applicant before fuel load. On the basis of its evaluation of the Perry pump and valve operability assurance program, the staff has identified the following confirmatory issues, which are being added to Section 1.11 of the SER by this supplement as License Condition (28):

(1) The applicant shall confirm that all of the required preoperational tests are completed before fuel load. At the time of the audit many of the preoperational tests for those systems required to be operational before fuel load had not been completed.

(2) The applicant shall confirm that all pumps and valves important to safety are qualified before fuel load. At the time of the audit all pumps and valves important to safety had not been qualified. For example, the applicant has indicated to the staff that qualification of the Dikkers safety/relief valves is scheduled to be completed in May 1985, shortly before the applicant's projected fuel load date of July 1985.

(3) The applicant shall confirm that the original loads used in tests and/or analyses to qualify pumps and valves important to safety are not exceeded by any new loads such as those imposed by a LOCA (hydrodynamic loads) or as-built conditions. If a new load exceeds that originally used, the impact of the new load on the qualification of the equipment must be assessed and reported to the NRC before fuel load.

3.11 Environmental Qualification of Electrical Equipment Important to Safety and Safety-Related Mechanical Equipment

3.11.1 Introduction

Equipment that is used to perform a necessary safety function must be demonstrated to be capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement - which is embodied in GDC 1 and 4 of Appendix A and Sections III, XI, and XVII of Appendix B to 10 CFR 50 - is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability for electrical equipment have been set forth in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"; NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," which supplements the Institute of Electrical and Electronics Engineers (IEEE) Standard 323; and various NRC regulatory guides and industry standards.

3.11.2 Background

NUREG-0588 was issued in December 1979 to promote a more orderly and systematic implementation of equipment qualification programs by industry and to provide guidance to the NRC staff for its use in ongoing licensing reviews.

The positions contained in that report provide guidance on (1) how to establish environmental service conditions, (2) how to select methods that are considered appropriate for qualifying equipment in different areas of the plant, and (3) other areas such as margin, aging, and documentation. In February 1980, the NRC asked certain near-term operating license (OL) applicants to review and evaluate the environmental qualification documentation for each item of safety-related electrical equipment and to identify the degree to which their qualification programs were in compliance with the staff positions discussed in NUREG-0588.

IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment," issued by the NRC Office of Inspection and Enforcement (IE) on January 14, 1980, and its supplements dated February 29, September 30, and October 24, 1980, established environmental qualification requirements for operating reactors. This bulletin and its supplements were provided to OL applicants for consideration in their reviews.

A final rule on environmental qualification of electrical equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, 10 CFR 50.49, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In conformance with 10 CFR 50.49, electrical equipment for Perry Units 1 and 2 may be qualified according to the criteria specified in Category 1 of NUREG-0588.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B of 10 CFR 50. The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment.

To document the degree to which the environmental qualification program complies with the NRC environmental qualification requirements and criteria, the applicant provided equipment qualification information by letters dated October 25, 1982, May 17, 1983, October 25, 1983, November 9, 1983, January 4, 1984, January 31, 1984, March 7, 1984, April 5, 1984, May 7, 1984, June 29, 1984, and October 2, 1984, to supplement the information in FSAR Section 3.11.

The staff has reviewed the adequacy of the Perry environmental qualification program for electrical equipment important to safety as defined in 10 CFR 50.49 and is in the process of reviewing the program for safety-related mechanical equipment. The scope of this report includes an evaluation of (1) the completeness of the list of systems and equipment to be qualified, (2) the criteria they must meet, (3) the environments in which they must function, and (4) the qualification documentation for the equipment. It is limited to electrical equipment important to safety within the scope of 10 CFR 50.49 and safety-related mechanical equipment.

3.11.3 Staff Evaluation

The staff evaluation included an onsite examination of equipment, an audit of qualification documentation, and a review of the applicant's submittals for completeness and acceptability of systems and components, qualification methods, and accident environments. The criteria described in SRP Section 3.11 (NUREG-0800) and in NUREG-0588, Category 1, and the requirements in 10 CFR 50.49 form the bases for the staff evaluation.

The staff performed an audit of the applicant's qualification documentation and installed electrical equipment on January 17, 18, and 19, 1984. The audit consisted of a review of 10 files containing information regarding equipment qualification. The staff's findings from the audit are discussed in Section 3.11.4.2 of this report.

3.11.3.1 Completeness of Equipment Important to Safety

10 CFR 50.49 identifies three categories of electrical equipment that must be qualified in accordance with the provisions of the rule. These are

(1) safety-related electrical equipment (equipment relied on to remain functional during and following design-basis events)

(2) non-safety-related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory accomplishment of the safety functions by the safety-related equipment

(3) certain postaccident monitoring equipment (RG 1.97, Rev. 2, Category 1 and 2 postaccident monitoring equipment)

The applicant has provided information addressing compliance with this requirement of 10 CFR 50.49.

The systems identified by the applicant for the environmental qualification program as being required to function to mitigate the consequences of loss-of-coolant accidents (LOCAs) or high-energy line breaks (HELBs) that have components located in a harsh environment were compared to FSAR Table 3.2-1, "Equipment Classification." The omission of systems from the harsh environment program was adequately justified by the applicant. Table 3.3 lists the systems identified and their safety functions.

To address conformance with 10 CFR 50.49(b)(2) concerning non-safety-related equipment whose failure under postulated accident conditions could prevent the satisfactory accomplishment of safety functions, the applicant stated that all such electrical equipment located in a harsh environment is classified as safety related. The applicant referred to compliance with IEEE Standard 384-1974 as modified by RG 1.75 to show electrical and physical separation between safety-related and non-safety-related electrical equipment. The staff has reviewed and evaluated the applicant's conformance with RG 1.75 and finds it acceptable from an equipment qualification aspect. The applicant has also conducted a detailed study in response to the concerns addressed by the staff in IE Information Notice 79-22, "Qualification of Control Systems," issued September 19, 1979. The staff has reviewed this study and finds it acceptable. Accordingly, the staff concludes that the applicant's conformance to 10 CFR 50.49(b)(2) is acceptable.

10 CFR 50.49(b)(3) requires that all installed RG 1.97, Revision 2, Category 1 and 2 instrumentation located in a harsh environment be included in the equipment qualification program unless adequate justification is provided. The applicant has indicated that all such equipment is included in the qualification program; however, in addressing conformance with RG 1.97, the applicant has identified a number of alternative methods of meeting the intent of RG 1.97. The staff will determine the acceptability of these alternative methods as part of its review for conformance with RG 1.97. This review may result in the addition of equipment to the environmental qualification program.

3.11.3.2 Qualification Methods

3.11.3.2.1 Electrical Equipment in a Harsh Environment

Detailed procedures for qualifying safety-related electrical equipment in a harsh environment are defined in NUREG-0588. The criteria in this NUREG report are also applicable to the other eqipment important to safety defined in 10 CFR 50.49.

The General Electric (GE) Environmental Qualification Program presented in GE Topical Report NEDE-24326-P (Rev. 1) outlines the methodology used by GE to qualify nuclear steam supply system (NSSS) safety-related electrical equipment subject to a harsh environment. The applicant, by letter dated February 22, 1983, adopted this GE program for Perry. The staff finds that the GE position on time margin, as presented in Topical Report NEDE-24326-P, does not address the requirement of NUREG-0588, which requires that time margin be a minimum of 1 hour. The staff considers this to be an open item in Topical Report NEDE-24326-P and requires that time margin be approached in accordance with NUREG-0588, or as amplified in RG 1.89. The applicant, by a letter dated October 2, 1984, has approached time margin in essentially the same manner as that specified in RG 1.89. The staff has reviewed the applicant's qualification methodology and finds it acceptable to meet the requirements of NUREG-0588, Category 1.

3.11.3.2.2 Safety-Related Mechanical Equipment in a Harsh Environment

Although there are no detailed requirements for mechanical equipment, GDC 1, "Quality Standards and Records," GDC 4, "Environmental and Missile Design Bases," and Appendix B to 10 CFR 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants" (Sections III, "Design Control," and XVII, "Quality Assurance Records"), contain the following requirements related to equipment qualifications:

(1) Components shall be designed to be compatible with the postulated environmental conditions, including those associated with LOCAs.

(2) Measures shall be established for the selection and review for suitability of application of materials, parts, and equipment that are essential to safety-related functions.

(3) Design control measures shall be established for verifying the adequacy of design.

(4) Equipment qualification records shall be maintained and shall include the results of tests and materials analyses.

The applicant has submitted a mechanical equipment qualification program for staff review. The review will determine if the environmental qualification of safety-related mechanical equipment has been adequately addressed. In addition, the staff will select three items from the program and conduct an indepth review of the documentation associated with these items. At this time the safety-related mechanical equipment qualification program review is incomplete.

3.11.3.3 Service Conditions

NUREG-0588 defines the methods to be used for determining the environmental conditions associated with LOCAs or HELBs, inside or outside containment. The review and evaluation of the adequacy of these environmental conditions are described below. The staff has reviewed the qualification documentation to ensure that the qualification conditions envelop the environmental conditions established by the applicant.

3.11.3.3.1 Temperature, Pressure, and Humidity Conditions Inside the Drywell

The applicant provided the LOCA/main steamline break (MSLB) profiles used for equipment qualification program submittals. The peak values in the drywell shown on these profiles are as follows:

	Maximum temperature, °F	Maximum pressure, psig	Humidity, %
LOCA/MSLB	330	22.1	100

The staff has reviewed these profiles and finds them acceptable for use in equipment qualification; that is, there is reasonable assurance that the actual pressures and temperatures will not exceed these profiles anywhere within the specified environmental zone (except in the break zone).

3.11.3.3.2 Temperature, Pressure, and Humidity Conditions Outside the Drywell

The applicant has provided the temperature, pressure, and humidity conditions associated with HELBs outside containment. The criteria used to define the location of HELBs are described in FSAR Section 3.6.

The staff has used a screening criterion of saturation temperature at the calculated pressure to verify that the peak temperatures identified by the applicant are acceptable.

The reactor water cleanup (RWCU) rooms in containment, but outside the drywell, are an exception to the screening criteria. The applicant has analyzed this exception and determined that no equipment in the RWCU rooms is needed for a break occurring in these rooms. The staff finds this acceptable.

3.11.3.3.3 Submergence

Flood levels for various areas have been calculated, with the flood level in the drywell being 599 ft following a LOCA. The effects of flooding on equipment have been evaluated to ensure that safe shutdown can be achieved. The applicant has taken appropriate corrective action to relocate or qualify all affected equipment.

3.11.3.3.4 Demineralized Water Spray

A demineralized water spray may be used inside primary containment to mitigate the effects of an accident. The effects of spray on equipment important to safety have been evaluated by the applicant.

3.11.3.3.5 Aging

The aging program requirements for Perry electrical equipment are defined in Category 1 of NUREG-0588. All degrading influences must be considered and included in the aging program. Justification for excluding pre-aging of equipment in type testing must be established on the basis of equipment design and application or state-of-the-art aging techniques. A qualified life is to be established for each equipment item.

In addition to the above, a maintenance/surveillance program must be implemented to identify and prevent significant age-related degradation of electrical and mechanical equipment. In the FSAR, the applicant committed to follow the recommendations in RG 1.33, Revision 2, "Quality Assurance Program Requirements (Operation)," which endorses American Nuclear Society/American National Standards Institute Standard ANS-3.2/ANSI N18.1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants." This standard defines the scope and content of a maintenance/surveillance program for safety-related equipment. Provisions for preventing or detecting age-related degradation in safety-grade equipment are specified and include (1) utilizing experience with similar equipment, (2) revising and updating the program as experience is gained with the equipment during the life of the plant, (3) reviewing and evaluating malfunctioning equipment and obtaining adequate replacement components, and (4) establishing surveillance tests and inspections based on reliability analyses, frequency, and type of service or age of the items, as appropriate.

The applicant has described a program that incorporates the above guidelines and has stated that the maintenance/surveillance program will be implemented at the time of fuel load. The specific program that will be used to detect unanticipated, age-related degradation of electrical cables inside and outside containment has been submitted. A staff review indicates that this program will adequately detect this degradation.

3.11.3.3.6 Radiation (Inside and Outside Containment)

The applicant has provided values of the radiation levels postulated to exist following a LOCA. The accident radiation environments in primary containment have been defined according to NUREG-0588. For this review, the staff has assumed that the values provided have been determined in accordance with the prescribed criteria. The staff review determined that the values to which the equipment is qualified enveloped the requirements identified by the applicant.

The maximum total radiation dose specified by the applicant for primary containment is 2.7×10^8 rads gamma. In the secondary containment, values of up to 4.1×10^7 rads gamma were used in the evaluation of equipment in areas exposed to recirculating fluid lines. These values are acceptable for use in the qualification of equipment.

3.11.3.4 Outstanding Equipment

For items not having complete qualification documentation, the applicant has provided a commitment for corrective action and a schedule for completion. For items that will not have full qualification before an operating license is granted, analyses must be performed in accordance with 10 CFR 50.49(i) to ensure that the plant can be operated safely pending completion of environmental qualification. These analyses must be submitted for consideration before the granting of an operating license.

3.11.4 Qualification of Equipment

The following subsections present the staff assessment based on the applicant's submittal, audit of documentation contained in the applicant's qualification files, information in the NRC Equipment Qualification Data Bank, and previous staff evaluations of equipment in other plants.

3.11.4.1 Electrical Equipment Important to Safety

The staff has separated the electrical equipment in a harsh environment into two categories: (1) equipment requiring additional qualification information or corrective action and (2) equipment considered acceptable pending implementation of the maintenance/surveillance program. Tables 3.4 and 3.5 give a listing of equipment in each of these categories.

3.11.4.1.1 Equipment Requiring Additional Information and/or Corrective Action

Table 3.4 identifies equipment in this category. Corrective action or deficiencies fall into the following categories and are noted in the table:

(1) QI - qualification information being developed
(2) R - radiation deficiency
(3) T - temperature deficiency
(4) TU - testing underway
(5) TNS - testing not started

The deficiencies have been determined on the basis of information available to the staff at the time of review and do not necessarily mean that the equipment is unqualified. However, the deficiencies are cause for concern and require further case-by-case evaluation. Before an operating license is issued, the applicant should review the qualification files to ensure that these deficiencies have been eliminated and the resolutions have been documented in an auditable form. The applicant must notify the staff that all equipment is qualified or must submit justification for interim operation in accordance with 10 CFR 50.49(i).

3.11.4.1.2 Equipment Considered Acceptable or Conditionally Acceptable

On the basis of the staff review, the items identified in Table 3.5 have been determined to be acceptable, pending implementation of the maintenance/ surveillance program.

3.11.4.2 Environmental Qualification Audit

The staff, with assistance from the Idaho National Engineering Laboratory, conducted an audit of the applicant's qualification files on January 17, 18, and 19, 1984. The purpose of the audit was to verify the bases of the information submitted by the applicant. Ten equipment qualification files, representing approximately 10% of the equipment items in the equipment qualification program, were selected for detailed review during the audit.

The equipment items selected for audit were

(1) terminal blocks, Buchanan Models NQB, NQO
(2) motor, Siemens-Allis Model 182T
(3) relays, Agastat Model E7012
(4) relays, Gould Model J10
(5) resistance temperature detector (RTD), WEED Model 611
(6) pressure transmitter, Rosemount Model 1153B
(7) electrical penetration, Westinghouse modular type
(8) differential pressure switch, Solon Model 7PS2DW
(9) motor operator, Limitorque Mode SMB-1-60
(10) solenoid valve, ASCO Model MP 8316

These files were reviewed to determine if qualification has been demonstrated on the basis of the documents contained in the files. Several deficiencies were noted and transmitted to the applicant by a letter dated February 14, 1984. The applicant responded by several docketed transmittals, which the staff reviewed. The staff concluded that the deficiencies have been adequately resolved.

As part of the audit, the equipment as actually installed was inspected during a plant walkdown. The purpose of the walkdown was to verify that the manufacturer, model number, location, and installation are consistent with qualification documents.

3.11.5 Conclusions

The staff has reviewed the Perry program for the environmental qualification of electrical equipment important to safety and safety-related mechanical equipment. The purpose of the review was to determine the adequacy of the program, including the scope of the qualification program, the environmental conditions resulting from design-basis accidents, and the methods used to demonstrate qualification.

As identified in this supplement, the following open items must be resolved:

(1) Before a license is issued, the applicant must notify the staff that all equipment is qualified or must submit justification for interim operation in accordance with 10 CFR 50.49(i) for all unqualified equipment.

(2) Before a license is issued, the applicant must certify that all mechanical equipment is qualified and submit three qualification files for staff review and approval, or provide justification for interim operation.

On the basis of the results of its review, and subject to the acceptable resolution of the open items identified above, the staff finds the Perry program for environmental qualification of electrical equipment important to safety and safety-related mechanical equipment acceptable. Consequently, SER Outstanding Issue (4), pertaining to environmental qualification, is being redefined accordingly in Section 1.9 of this supplement.

Table 3.1 Equipment audited

SQRT ID No.	Applicant ID No	Equipment name and description	Safety function	Findings	Resolution	Status	Remarks
BOP-1	10P47C0001B	Control complex chilled water pump	Supplies cooling water for control room chillers			Qualified	
BOP-2	1R42S0003	Battery and rack, lead acid electrical storage batteries and rack that supports the batteries	Provides 225 Vdc for the safety systems			Qualified	
BOP-3	11M51F0010B	4-in., 300-lb motor-operated globe valve	Provides isolation when shut and throttling when open for the combustible gas control system			Qualified	
BOP-4	1D17F0079A	1-in. solenoid valve	Provides containment isolation			Qualified	
BOP-5	0H51P0077A	Emergency service water traveling water screen control panel	Controls the traveling screen			Qualified	
BOP-6	1R24S0032	Motor control center	Provides 480 Vdc and 125 Vdc to 1E electrical equipment			Qualified	
BOP-7	1P45C0002	Vertical pump, 41.75 ft long by 11-3/8-in. diameter with drive motor mounted on top	Supplies water to the emergency service water system			Qualified	
BOP-8	1M39B0004	Air handling unit	Provides ventilation and cooling air			Qualified	

Table 3.1 (Continued)

SQRT ID No.	Applicant ID No.	Equipment name and description	Safety function	Findings	Resolution	Status	Remarks
BOP-9	1M51S0002	Power supply panel, electrical cabinet with transformers, magnetic contactor, and switch control rectifier panel mounted inside	Supplies power to a hydrogen recombiner			Qualified	
BOP-10	0C41N0415B	Level switch for instrument with dial	Is a part of the standby liquid control system (auxiliary mixing tank level indicator)			Qualified	
BOP-11	1R72S0002	Medium voltage modular electrical penetration	Maintains the containment pressure boundary			Qualified	
NSSS-1	1C71S0003G	Electrical protection assembly	Performs a 1E protection function			Qualified	
NSSS-2	1H22P0072	Rod position multiplexer cabinet, cabinet containing instruments	Transmits correct rod position	The question of whether this equipment is safety related or not is being investigated by General Electric	Pending	Open	
NSSS-3	1C11F0180	Control rod drive scram discharge volume vent valve	Normally open and required to close when a scram occurs			Qualified	
NSSS-4	1C41F0004B	Squib valve, 4.5 in. by 7-in. diameter, electrically triggered explosive valve	Provides an injection pathway for poison into the primary coolant for the standby liquid control system			Qualified	

Table 3.1 (Continued)

SQRT ID No.	Applicant ID No.	Equipment name and description	Safety function	Findings	Resolution	Status	Remarks
NSSS-5	1E51C0002	Reactor core isolation cooling (RCIC) steam turbine assembly	Drives the RCIC pump providing water to the reactor vessel to maintain inventory during reactor isolation, the control rod drop accidents, and anticipated transients without scram			Qualified	
NSSS-6	1H13P0702	Termination cabinet, electrical cabinet containing cable termination hardware	Provides an interface between control room circuitry and electrical cables entering the control room			Qualified	
NSSS-7	1H13P0680	Control room panel, a panel containing instruments	Is a principal plant control console			Qualified	
NSSS-8	1H13P0870	Benchboard, electrical panel with displays, indicators and switches mounted on it	Provides support for electrical equipment such as displays, indicators, and switches			Qualified	
NSSS-9	0G41F0085	10-in. butterfly valve	Open, controls flow; closed, shuts off flow in the fuel pool return line			Qualified	

Table 3.1 (Continued)

SQRT ID No.	Applicant ID No.	Equipment name and description	Safety function	Findings	Resolution	Status	Remarks
NSSS-10	2E12N0057	Transmitter, differential pressure	Measures water pressure in the residual heat removal system shutdown cooling suction piping and provides an electrical output signal proportional to the measured pressure			Qualified	
		Control room ceiling	Provides control room lighting. Failure of the ceiling could damage the equipment in the control room and injure operators			Qualified	Not a Seismic Qualification Review Team list item

Table 3.2 Summary of audit by Pump and Valve Operability Review Team

Plant ID No.	Description	Safety function	Findings	Resolution	Status	Remarks
E51-C001	Reactor core isolation cooling pump (NSSS)	Supplies makeup water to reactor vessel during loss of feedwater	a	b	Closed[c]	None
E21-C001	Low-pressure core spray pump (NSSS)	Supplies makeup water and heat removal	a	b	Closed[c]	None
E22-F0010	High-pressure core spray (HPCS) valve to condensate storage (NSSS)	Closed, it isolates bypass to condensate storage	a	b	Closed[c]	None
P47-C001	Control complex chill water water pump (BOP)	Supplies chilled water to control complex	a	b	Closed[c]	None
M51-F0110	Combustible gas control valve (BOP)	Closed, it isolates backup hydrogen purge line	a	b	Closed[c]	None
E12-F0048	Residual heat removal system heat removal valve (BOP)	Open, it provides low-pressure coolant injection and suppression pool cooling	a	b	Closed[c]	None
E22-C003	HPCS water leg pump (BOP)	Pressurizes HPCS pump discharge line	a	b	Closed[c]	None
	All pumps and valves important to safety		a,d,e	b,f,g	Closed[c]	None

[a] The applicant has not completed all required preoperational tests.

[b] The applicant shall confirm that all pumps and valves important to safety have had their required preoperational tests completed before fuel load.

[c] The qualification status is considered closed, pending completion of the resolutions.

[d] The applicant has not completed the qualification of all pumps and valves important to safety.

[e] The applicant has not verified that all new loads are enveloped by those loads originally used to qualify the equipment.

[f] The applicant shall confirm that all pumps and valves important to safety are qualified before fuel load.

[g] The applicant shall confirm that none of the new loads applicable to pumps and valves important to safety exceed those loads originally used to qualify the equipment.

NOTE: NSSS = nuclear steam supply system; BOP = balance of plant.

Table 3.3 Safety-related systems - Perry environmental qualification program

System name	Safety function*
Airborne effluent	
Annulus exhaust gas treatment	A,B,C,D,E,F
Auxiliary ac power	A,B,C,D,E,F
Containment isolation valves	B
Control room panels	A,B,C,D,E,F
Control rod drive hydraulic	A,F
Diesel generator	A,B,C,D,E,F
Emergency closed cooling water	D,E,F
Emergency service water	D,E,F
Feedwater leakage control	A,B,C,E,F
Fuel pool cooling and cleanup	B,D,F
Heating, cooling and ventilation	A,B,C,D,E,F
High-pressure core spray	A,B,C,E,F
Heating, ventilating, and air conditioning (HVA) critical area cooling	A,C,D,E
HVA piping and isolation	A,B,F
Instrument and service air	A,B
Local panels and racks	A,B,C,D,E,F
Low-pressure core spray	A,B,C,E,F
Main steamline isolation valve leakage control	B,F
Neutron monitoring	A
Nuclear boiler	A,B,C,E,F
Plant service and cooling water	B

*See footnote at end of table.

Table 3.3 (Continued)

System name	Safety function*
Primary containment area monitors	
Process radiation monitors	A,B,F
Radwaste	B,F
Reactor core isolation cooling	A,B,C,E,F
Recirculation	
Reactor protection system	A
Reactor water cleanup	B,C
Residual heat removal	A,B,C,D,E,F
Secondary containment area monitors	
Standby liquid control	A,F
Suppression pool makeup	A,E,
125-Vdc power	A,B,C,D,E,F

*A--emergency reactor shutdown
B--containment isolation
C--reactor core cooling
D--containment heat removal
E--reactor heat removal
F--effluent control

Table 3.4 Equipment requiring additional information or corrective action

Component	Manufacturer	Model number	Deficiency/ corrective action*
BALANCE-OF-PLANT SCOPE			
Electrohydraulic actuator	ITT General Controls (Bettis)	INH95	QI
Differential pressure switch	Barton	580 A	QI
Level transmitter	Gould	PD 3218	QI
Radiation monitor	KAMAN	KDA-HR	QI
Radiation monitor	KAMAN	KMG-HRH	QI
Hydrogen igniter	Power Systems Inc.	6043	QI, R,T
Heater and controls	CVI Penwalt Corp.	#B 435-6011 #B 435-6014	QI
Limit switches	NAMCO	EA740, 170, 180	TU
Multicable	Nelson Electric	RSG-6	TNS
Distribution panel	Not selected	-	-
Power lead gland	Conax	Various	TNS
Motor	Reliance	365T	QI
NUCLEAR STEAM SUPPLY SYSTEM SCOPE			
Safety relief valve	Dikkers	G471-6/125.04	TNS
Main steam isolation valve actuator	Sheffer	SA-A34	TNS
Limit switch	NAMCO	EA740-50100 REV K	TNS

*See footnote at end of table.

Table 3.4 (Continued)

Component	Manufacturer	Model number	Deficiency/ corrective action*
NUCLEAR STEAM SUPPLY SYSTEM SCOPE (Continued)			
Main steam isolation heater	General Electric (LOMPAC)	47D518673	QI
Blower motor	General Electric (LOMPAC - SEIMENS)	47D518673	QI
Reactor core isolation cooling steam turbine assembly	Terry Corp.	GS-2N	QI
Pressure transmitter	Rosemount	1152 1151/T0280	TNS
Temperature elements	PYCO	102-9039-11	TNS
Flow meter	S and K	20-9651-8550	TNS
Flow transmitter	S and K	91X-16-4-20	TNS
Pressure transmitter	Rosemount	1153	TNS
Pressure switch	Pressure Control Inc.	A17-1P	QI
Solenoid valve	Valcor	V70900-43 V70900-45 V70900-46	QI
Level switch	Magnetrol	5.0-751-1X-MPG -M14HY	QI
Level transmitter	Gould	PD3218	QI
Power range monitor connector	Amphenol	X901-199	TNS

*QI--qualification information being developed
R--radiation deficiency
T--temperature deficiency
TU--testing underway
TNS--testing not started

Table 3.5 Equipment considered qualified pending implementation of surveillance and maintenance program

Component	Manufacturer	Model number
BALANCE-OF-PLANT SCOPE		
Motor connection splice	Raychem	N-MCK
Cable splice assembly	Raychem	WSF-N
Cable breakout and end sealing kit	Raychem	NCBK, NESK
Motor connection splice	Raychem	NMCK8
Splice connection kit	Raychem	NPKV
High voltage termination kit	Raychem	NHVT
Terminal blocks	Buchanan	NBQ, NBO
Lug connectors	Burndy	YAES-K
Differential pressure switch	Solon	7PS2DW
Motor	Siemens-Allis	182T
Power cable	Anaconda	Uniblend EP
Power and control cable, chemically cross-length polyethylene (XLPE)	Rockbestos	Firewall III
Power and control cable, irradiation XLPE	Rockbestos	Firewall III
Instrument cable	Brand-Rex	XLPE
Electrical penetrations	Westinghouse	Modular
Thermocouple cable	Samuel Moore	Multi-Pair 16 AWG
Valve motor operator	Limitorque	SMB, SB SMB/HBC
Indicating light	General Electric	ET-16
Control switch	General Electric	CR2940
Terminal boards	General Electric	CR151B

Table 3.5 (Continued)

Component	Manufacturer	Model number
BALANCE-OF-PLANT SCOPE (Continued)		
Wire	General Electric	SP-57279
Terminal/fuse blocks	Buchanan	NQ0211 NQ0361
Relays	Agastat	7012
Relays	Gould	J10
Terminals	Thomas and Betts	STA-KON RBT853
Resistance temperature detector (RTD)	WEED	611
RTD	WEED	612
Temperature transmitter	WEED	4000R
Thermocouple	WEED	E4D250G -7A1
Solenoid valve with position switch	Target Rock	77JJ series
Pressure transmitter	Rosemount	1153
Solenoid valve	ASCO	NP 8320
Hydrogen recombiner	Westinghouse	A
Motors	Reliance	Type P 405 TS Frame
Solenoid valve	ASCO	NP 8316
Motors	Reliance	324T, 254T 405TX, 405T 405TS, 326TS 215T
Motors	Reliance	256T, 184T
Coaxial cable	Rockbestos	Firewall III
Instrument cable	Rockbestos	Firewall III
Seal assembly	Conax	Various

Table 3.5 (Continued)

Component	Manufacturer	Model number
NUCLEAR STEAM SUPPLY SYSTEM SCOPE		
Motors	General Electric	5K6338XC105A 5K6347XC100A 5K6357XC18A
Explosive valve	Conax	1832-159-01 1532-159-01
Hydraulic control units	General Electric/ASCO	767E8006001 HVA-176-186-1

4 REACTOR

4.2 Fuel System Design

4.2.1 Design Bases

4.2.1.2 Fuel Rod Failure Criteria

(8) Fuel Rod Mechanical Fracturing

The term "mechanical fraturing" refers to a cladding defect that is caused by an externally applied force, such as a hydraulic load or load derived from core plate motion. Such loads are bounded by the loads of a LOCA and safe shutdown earthquake (SSE), and a mechanical fraturing analysis is usually performed as a part of the seismic-and-LOCA loads analysis (see Section 4.2.1.3(4) of this supplement). General Electric (GE) has described the seismic-and-LOCA analysis, which includes fuel rod fracturing, in the staff-approved GE Topical Report NEDE-21175-P, Amendment 3 (NEDE-21175-P-3).

In Section 4.2.1.2(8) of the SER, the staff reported, as a confirmatory issue, the need for the applicant to perform such a fuel rod mechanical fracturing analysis for Perry. In updating the status of this issue in SSER No. 4, the staff clarified that the applicant provide such a plant-specific analysis using the accepted GE generic methodology described in NEDE-21175-P-3 to fully resolve this issue. By letter dated August 24, 1984, the applicant indicated that the BWR design limits for Perry are contained in NEDE-21175-P-3. The staff has confirmed that the fuel rod mechanical fracturing design limits for Perry fuel design are consistent with the criteria in NEDE-21175-P-3. The staff's evaluation findings are presented in Section 4.2.3.2 of this supplement.

4.2.1.3 Fuel Coolability Criteria

(4) Fuel Assembly Structural Damage From External Sources

Earthquakes and postulated pipe breaks in the reactor coolant system could result in external forces acting on fuel assemblies. Section 4.2.1 of the Standard Review Plan (NUREG-0800) and Appendix A to that section state that fuel system coolability should be maintained and that any damage (including liftoff) whould not be so severe as to prevent control rod insertion when it is required during these low-probability accidents. GE has described the seismic-and-LOCA loads analysis in NEDE-21175-P-3.

In Section 4.2.1.3(4) of the SER, the staff reported that, as a further confirmatory issue, the applicant was required to perform a plant-specific analysis of fuel assembly structural damage from external sources, using the staff-approved generic methodology described in NEDE-21175-P-3. In SSER No. 4, the staff clarified that this analysis be performed in conjunction with the fuel mechanical fracturing analysis (see Section 4.2.1.2(8) above). In a letter dated August 24, 1984, the applicant indicated that the design limits for BWR fuel contained in NEDE-21175-P-3 were applicable to Perry. The staff has confirmed this, and its evaluation findings are presented in Section 4.2.3.3 of this supplement.

4.2.3 Design Evaluation

4.2.3.2 Fuel Rod Failure Evaluation

(8) Fuel Rod Mechanical Fracturing

As stated in Section 4.2.1.2(8) above, the applicant has referenced the analysis and generic methodology described in NEDE-21175-P-3 as the plant-specific analysis for Perry. The staff finds that the analytical results contained in NEDE-21175-P-3 satisfy its requirement and considers Confirmatory Issue (11), listed in Section 1.10 of the SER, resolved. Since the mechanical fracturing analysis is usually accomplished as part of the seismic-and-LOCA loads analysis, further discussion on which Confirmatory Issue (11) is considered resolved is presented in Section 4.2.3.3(4) of this supplement.

4.2.3.3 Fuel Coolability Evaluation

(4) Fuel Assembly Structural Damage From External Sources

As stated in Section 4.2.1.2 of SSER No. 4 and in this supplement, the staff has approved the generic methods for analyzing fuel damage described in NEDE-21175-P-3, that is, the method for evaluating seismic-and-LOCA loads. The staff has reviewed the plant-specific values for liftoff and acceleration presented in the applicant's letter dated August 24, 1984, which are predicated on the approved GE methodology. The results show the vertical liftoff for Perry fuel assemblies is less than the allowable limit given in NEDE-21175-P-3, and the accelerations are also within the evaluation-basis limits, thereby ensuring the structural integrity and control rod insertability of the Perry fuel assembly design during seismic-and-LOCA events. Consequently, Confirmatory Issue (12), listed in Section 1.10 of the SER, is also considered to be satisfactorily resolved.

4.4 Thermal-Hydraulic Design

4.4.7 TMI-2 Action Plan Item II.F.2

4.4.7.1 Inadequate Core Cooling (ICC) Detection System

In Section 4.4.7.1 of SSER No. 4, the staff reported that it would complete its review of GE Reports SLI-8211 and SLI-8218 submitted by the BWR Owners Group pertaining to SER License Condition (4) concerning instrumentation for detecting inadequate core cooling. It was indicated that any additional instrumentation requirements for incore thermocouples, resulting from the development of a final staff position relative to its review of the GE reports, would be appropriately identified. In SSER No. 4, SER License Condition (4) was accordingly modified as follows:

> The applicant shall implement the staff's requirements regarding additional instrumentation for the detection of inadequate core cooling per TMI Action Plan Item II.F.2, based on the staff's review

of BWR Owners Group [GE] Reports SLI-8211 and SLI-8218, and the applicant's plant-specific evaluation report addressing the recommendations of the BWR Owners Group prior to fuel load.

The staff has since completed its review of GE Reports SLI-8211 and SLI-8218 and the applicant's plant-specific responses contained in letters dated January 14, 1983, November 1, 1983, and January 14, 1985, which address the three water level instrumentation concerns expressed in GE Report SLI-8211. The staff finds the applicant's responses acceptable and concludes that the Perry instrumentation acceptably conforms to the recommendations cited in GE Reports SLI-8211 and SLI-8218, thus permitting the above cited license condition to be withdrawn.

The staff's position on the issue of inadequate core cooling detection is that if the applicant upgrades the water level system to be consistent with the recommendation of Report SLI-8211, then there is no additional instrumentation required for inadequate core cooling detection.

Previously, as specified in Regulatory Guide 1.97, Rev. 2 (December 1980), incore thermocouples were to be installed in BWRs. However, in 1981, the Advisory Committee on Reactor Safeguards (ACRS) recommended (as stated on page 2 of ACRS Report No. 0938, dated August 11, 1981) that the incore thermocouple requirement be reevaluated. In addition, the BWR Owners Group submitted a report to the staff (as Appendix B of Report SLI-8218, dated December 1982) which concluded that the effectiveness of incore thermocouples as indicators of inadequate core cooling is very limited and accordingly recommended that incore thermocouples not be used to detect inadequate core cooling. Subsequently, Supplement No. 1 to NUREG-0737 (December 1982) deleted the requirement for incore thermocouples.

The staff, in reviewing the BWR Owners Group recommendation, questioned the reliability of existing water level instrumentation as the sole indication of inadequate core cooling and requested that a further study be performed by the Owners Group to evaluate the need for upgrading existing water level instrumentation to make it more reliable as a detector of inadequate core cooling. The staff also requested that the Owners Group consider what other instrumentation might be needed in the plant monitoring system for BWRs. To reflect this review status, RG 1.97, Revision 3 (May 1983) deleted the provision for installation of incore thermocouples; that is, the staff provided BWR applicants an opportunity to demonstrate that available means of detecting inadequate core cooling other than the installation of incore thermocouples are adequate.

In response, the BWR Owners Group submitted the following two GE reports for staff review and approval;

(1) Report SLI-8211 (July 1982), "Review of Reactor Water Level Measurement Systems," which includes the Owners Group's evaluation of existing water level instruments and recommendations for their improvement.

(2) Report SLI-8218 (December 1982), "Inadequate Core Cooling Detection in BWRs," which presents evaluation results of additional instrumentation as diverse indicators of inadequate core cooling and recommendations regarding the need for such additional instrumentation for BWR plant monitoring systems.

At the staff's request, the applicant also submitted a plant-specific evaluation (letters dated January 14, 1983, November 1, 1983, and January 14, 1985) addressing the applicability of the BWR Owners Group findings (GE Reports SLI-8211 and SLI-8218) to Perry.

The staff has completed its review of Report SLI-8211, and the results of this review are included in NRC Generic Letter 84-23 (October 26, 1984). The staff has also reviewed the applicant's responses describing modifications to the water level measurement systems at Perry to make them more reliable during postulated accident conditions. These modifications include (1) rerouting of instrument sensing lines within the drywell to limit the overall vertical drop to within 30 in., (2) reduction of the differences in drops between associated reference and variable legs to less than 26 in., and (3) the relocation of the instrument line flow limiting orifice plates nearer to the corresponding drywell penetration. The applicant also stated that analog trip units, rather than the less reliable mechanical types, are used at Perry and that the Perry protection logic design (for reactor trip and/or engineered safety feature system actuation on reactor vessel low water level) has four divisions and is identical to the "plant B" design described in Report SLI-8211. In its review of the "plant B" design, the staff found no cases identified which failed to provide automatic reactor trip and emergency core cooling system actuation. The staff concludes that no changes are required for the Perry protection system logic and that the Perry water level measurement system is in compliance with the BWR Owners Group recommendations in Report SLI-8211 and RG 1.97, Revision 3. As such, the Perry design is considered to be acceptable.

The staff has also completed its review of Report SLI-8218 and agrees with the conclusions delineated therein that the application of both additional inadequate core cooling devices and water level measurement reliability improvements is not justified by the resulting risk reduction. The risk remaining after inclusion of the water level measurement reliability improvements cited in Report SLI-8218 is sufficiently small (on an absolute basis) to preclude the need for further reduction in risk that would be obtained through the use of additional inadequate core cooling devices. Therefore, the staff agrees with the conclusion drawn in Report SLI-8218 that if the applicant upgrades the water level system to be consistent with the recommendations cited in Report SLI-8211, additional instrumentation is not needed for detection of inadequate core cooling. Since the Perry water level instrumentation conforms with the recommendations of Report SLI-8211, no additional instrumentation is required for detection of inadequate core cooling. The staff concludes that the license condition regarding the requirements for detection of inadequate core cooling may be removed. Consequently, SER License Condition (4) is resolved, is no longer considered relevant, and is being deleted by this supplement in view of the above evaluation findings and conclusions.

5 REACTOR COOLANT SYSTEM

5.2 Compliance With Code and Code Cases

5.2.5 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This section was prepared with the technical assistance of U.S. Department of Energy (DOE) contractors from the Idaho National Engineering Laboratory.

This evaluation supplements conclusions in Section 5.2.5 of the SER, which addresses the definition of examination requirements and the evaluation of the applicant's compliance with 10 CFR 50.55a(g).

5.2.5.2 Evaluation of Compliance of Perry Unit 1 With 10 CFR 50.55a(g)

The staff has completed its review of the information presented in the FSAR through Amendment 14 dated August 1984, the Perry Unit 1 preservice inspection (PSI) program including revisions through October 1984, and other related correspondence submitted by the applicant on or before December 14, 1984. The PSI program was essentially a completed document when the section containing the reactor pressure vessel (RPV) was submitted by the applicant on August 28, 1984. The scope of the examinations, procedures, and acceptance criteria are based on the requirements of ASME Code, Section XI, 1977 Edition with Addenda through Summer 1978. The extent of examination has been determined by the requirements of ASME Code, Section XI, 1974 Edition with Addenda through Summer 1975. The staff has concluded that the PSI program for systems and components within the reactor coolant pressure boundary is consistent with the applicable Code requirements except as discussed in the following paragraphs.

The staff review of the August 28, 1984, submittal determined that the selection of welds for the RPV examination is consistent with the applicable Code requirements. However, Paragraph 2.0 of the RPV PSI program states in part that "consideration will be given to the provisions of Regulatory Guide 1.150." The applicant also stated in a submittal dated March 25, 1982, that the provisions of RG 1.150 will be addressed and will be reflected in the ultrasonic examination procedures. Since the available information does not address the degree of compliance with RG 1.150, the applicant should document for staff review the measures taken to meet this regulatory guide.

At a public meeting on March 9, 1982, the applicant was requested to provide justification for the exclusion from examination of pipe with an inside diameter (ID) of 2.36 in. for water service and 4.72 in. for steam service on the basis of normal makeup for Class 1 systems. Exclusion of pipe welds greater than 3.0 in. in nominal diameter based on normal makeup flow is not consistent with the applicable Paragraph IWB-1220 of Section XI of the Code. The applicant responded in a letter dated March 25, 1982, that an exclusion from examination of 3.0 in. ID piping would be requested instead of 4.72-in. ID piping as listed in the PSI program. Revision 3 of the PSI Program - General Reference Text dated April 26, 1984, changed the exclusion diameter from 4.72 to 3.0 in. ID. However, the December 14, 1984, submittal of Revision 2 of the main steam system section of the PSI program still allows the exclusion diameter of 4.72 in. ID. This section should be revised to be consistent with the above commitment.

In the applicant's submittal dated December 14, 1984, the PSI program includes a revision to Document Number 80A4428 entitled "Reactor Recirculation System," which identifies approximately 44 welds that are clad. In a letter dated July 2, 1981, the applicant responded to Generic Letter 81-03 and indicated that corrosion-resistant cladding followed by solution annealing was applied to the piping in the recirculation system to minimize the problem of intergranular stress corrosion cracking. The letter also stated that all NSSS-supplied stainless steel piping now conforms to the corrosion-resistant material requirements of Section III of NUREG-0313, Revision 1. The staff's review of the applicant's letter was presented in SER Section 5.2.4.1, "Stainless Steel Pipe Cracking." Table 2.4 of Document Number 80A4428 lists four calibration standards for 24-in., 22-in., 16-in., and 12-in. diameter clad pipe and states that design drawings will be presented at a later date. Corrosion-resistant cladding can influence the ability to perform an effective ultrasonic examination with conventional instrumentation, especially where weld metal was applied to the outside diameter of the pipe for weld shrinkage or concentricity reasons. Therefore, the staff requests that the applicant provide the design drawings for the cladding calibration standards listed in Table 2.4 and confirm that the cladding thickness and width in the drawings are representative of the dimensions of the cladding applied to the recirculation piping. The staff also requests that the applicant address the measures that were taken to ensure that the instrumentation and examination procedures used on the clad welds were capable of detecting a significant flaw, if present.

The specific areas where the Code requirements cannot be met will be identified after the examinations are performed. The applicant has committed to identify all plant-specific areas where the Code requirements cannot be met and provide a supporting technical justification.

The staff still considers the review of the PSI program a confirmatory issue contingent on the applicant providing the following information:

(1) Address the degree of compliance with RG 1.150.

(2) Revise the main steam system section to delete reference to the exclusion diameter of up to 4.72 in. ID for steam service based on normal makeup for Class 1 systems.

(3) Address the ultrasonic examination of welds with corrosion-resistant cladding.

(4) Submit all relief requests with a supporting technical justification.

The staff will complete the evaluation of the PSI program in a future supplement after the applicant provides an acceptable response. Therefore, SER Confirmatory Issue (15) continues to remain unresolved.

The initial inservice inspection program has not been submitted by the applicant. The program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b) but before the first refueling outage when inservice inspection commences.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.8 Fracture Prevention of Containment Pressure Boundary

In Section 6.2.8 of the SER, the staff concluded from its review of the FSAR that there was reasonable assurance that brittle fracture of the Perry containment pressure boundary materials would not occur; however, to verify this conclusion, the staff required the applicant to provide specific confirmatory information demonstrating that the lowest temperature that would be experienced by the limiting materials under the conditions cited in GDC 51 agreed with the limiting temperatures identified by staff analysis. The need for this confirmatory information was inadvertently omitted from the list of issues in Section 1.10 of the SER, and since the required information has not been provided by the applicant, it is being added as Confirmatory Issue (58) by this supplement.

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of DOE contractors from the Idaho National Engineering Laboratory.

This evaluation supplements conclusions in Section 6.6 of the SER, which addresses the definition of examination requirements and the evaluation of the applicant's compliance with 10 CFR 50.55a(g).

6.6.3 Compliance of Perry Unit 1 With 10 CFR 50.55a(g)

The staff has completed its review of the information presented in the FSAR through Amendment 14 dated August 1984, the Perry Unit 1 preservice inspection (PSI) program including revisions through October 1984, and other related correspondence submitted by the applicant on or before December 14, 1984. The scope of the examinations, procedures, and acceptance criteria is based on the requirements of ASME Code, Section XI, 1977 Edition with Addenda through Summer 1978. The extent of examination has been determined by the requirements of ASME Code, Section XI, 1974 Edition with Addenda through Summer 1975. The staff has reached the conclusion that the PSI program for Class 2 and 3 systems and components is consistent with the applicable Code requirements.

The specific areas where the Code requirements cannot be met will be identified after the PSI examinations are performed. The applicant has committed to identify all plant-specific areas where the Code requirements cannot be met and provide supporting technical justification.

The staff has determined that the PSI program for Class 2 and 3 components is acceptable and that the review is considered to be a confirmatory issue contingent on the applicant submitting all relief requests with a supporting technical justification. The staff will complete the evaluation of the PSI program in a future supplement after the applicant provides an acceptable response.

The initial inservice inspection program has not been submitted by the applicant. The program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b) but before the first refueling outage when inservice inspection commences.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Protection System

7.2.2 Specific Findings

7.2.2.8 Instrumentation Setpoints

During the operating license review of the Perry Technical Specifications, the staff identified a concern regarding the values selected for protection system instrument setpoints and, in general, the methodology used to establish the reactor protection system setpoints. During the staff's review, it was determined that additional information would be required to confirm the applicant's conformance with the Commission's regulations relevant to the issue of protection system setpoints. The applicable regulations are GDC 20, 10 CFR 50.36, and 10 CFR 50.46. Guidance on acceptable methods for complying with these regulations is contained in RG 1.105, "Instrumentation Setpoints."

In an effort to conserve resources while providing the requested information, the applicant joined with several other BWR owners to form the Licensing Review Group (LRG) - Instrumentation Setpoint Methodology Group (ISMG). On July 14, 1983, the staff met with the ISMG at their request. At this meeting the ISMG presented an outline of a setpoint methodology. In response to additional questions from the staff, another meeting was held on January 31, 1984. By letter dated May 15, 1984 (T. M. Novak to J. F. Carolan (Chairman, ISMG)), the staff provided its assessment of the ISMG methodology. The staff evaluation identified several deficiencies in the methodology presented and requested that the ISMG provide additional information in response to 10 specific concerns. In response to the staff's evaluation, by letter dated June 29, 1984 (J. F. Carolan to T. M. Novak), the ISMG provided an action plan for resolving the outstanding issues. By letter dated July 23, 1984 (B. J. Youngblood to J. F. Carolan), the staff accepted the proposed action plan, and by letter dated October 9, 1984, the applicant committed to the work scope and schedule proposed by the ISMG action plan. The final acceptability of the protection system instrumentation setpoints will be addressed following completion of the staff's review of the forthcoming additional information.

The staff concludes that there is reasonable assurance, based on staff participation in meetings with the ISMG, that the forthcoming more detailed information on the setpoint methodology being developed by this group will verify the acceptability of the proposed setpoints. In the interim, the staff finds the setpoints contained in the Perry Technical Specifications acceptable.

7.3 Engineered Safety Features Systems

7.3.2 Specific Findings

7.3.2.7 Manual Initiation and Termination of ESF Systems

In Section 7.3.2.7 of SSER No. 4, the manual initiation and termination of engineered safety feature (ESF) systems (originally identified as Outstanding

Issue (12) in the SER) was changed to Confirmatory Issue (55). In order for the staff to consider this issue fully resolved, the applicant was required to provide written confirmation that the labeling for the low-pressure coolant injection (LPCI) and low-pressure core spray (LPCS) injection valves in the control room will be appropriately changed to reflect a warning that inadvertent actuation could cause overpressurization of the LPCI or LPCS piping. These labels were to be designed, fabricated, and located in accordance with good human factors and engineering principles and be installed in the control room before fuel load.

By letter dated January 21, 1984, the applicant stated that a blue indicating light will be installed above the control switches for LPCI and LPCS injection valves E12-F012A, B, and C and E21-F005. The light will go on to alert the operator that the reactor coolant pressure is low enough to prevent overpressurization of the LPCI or LPCS piping when the injection valves are opened. The use of the blue light for permissive indication is consistent with human factors standards employed at Perry. The applicant also stated that operator training will address the above modification.

On the basis of the information furnished by the applicant in the January 21, 1984, letter, the staff finds that the means proposed for protecting the LPCI/LPCS piping and warning the operator that opening of these system injection valves may cause overpressurization of the LPCI or LPCS piping are acceptable. Accordingly, Confirmatory Issue (55), introduced in SSER No. 4, is considered to be resolved.

7.5 Safety-Related Display Instrumentation

7.5.2 Specific Findings

7.5.2.2 Conformance to Regulatory Guide 1.97, Revision 2

It is stated in Section 7.5.2.2 of the SER that the applicant committed to upgrade postaccident monitoring instrumentation in accordance with the guidance of RG 1.97, Revision 2, "Instrumentation for Light Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident." It is further stated that the staff will review the upgraded postaccident monitoring provisions at Perry on a schedule consistent with forthcoming implementation requirements.

RG 1.97, Revision 2, implementation requirements were communicated to the applicant by NRC Generic Letter 82-33 (December 17, 1982). The applicant responded to Generic Letter 82-33 by letter dated July 19, 1983, which transmitted Amendment 12 to the Perry FSAR. FSAR Amendment 12 identified deviations proposed to RG 1.97, Revision 2, variables, including neutron flux, drywell sump level, drywell drain sump level, radiation level in circulating primary coolant, suppression chamber spray flow, standby liquid control system flow, residual heat removal heat exchanger outlet temperature, cooling water temperature to ESF system components, and main steamline isolation valve leakage control system pressure.

The staff and its consultant, EG&G Idaho, Inc. (EG&G), have reviewed the above noted deviation items. The EG&G interim report was submitted to the applicant by NRC letter dated December 11, 1984 and is being added to the SER as Appendix J by this supplement. The interim report concludes that Perry either conforms with or is justified in deviating from the guidelines of RG 1.97, Revision 2, with the following exceptions:

(1) Neutron flux - The present Perry instrumentation for this variable is acceptable on an interim basis until Category I instrumentation is developed and installed. The applicant is required to advise when Category 2 instrumentation will be installed in the plant (see Section 3.3.1 of Appendix J).

(2) Suppression chamber spray flow - The applicant must show that the instrumentation for this variable is in conformance with RG 1.97, Revision 2 (see Section 3.3.4 of Appendix J).

(3) Residual heat removal heat exchanger outlet temperature - The applicant needs to provide additional information on the instrumentation for this variable (see Section 3.3.7 of Appendix J).

(4) Cooling water temperature to ESF system components - The applicant needs to provide a justification basis for the lower limit of emergency closed cooling with temperature being 50°F (see Section 3.3.8 of Appendix J).

The applicant has been requested in the December 11, 1984, letter to respond to the above unresolved variables by February 1985. This information is needed to enable the staff and its consultant to complete their review of the proposed deviations to RG 1.97, Revision 2. License Condition (29) is being added to Section 1.11 of the SER by this supplement requiring that acceptability of all deviations be determined by the staff before fuel loading, and further, that all plant instrumentation system modifications required to comply with the guidelines of RG 1.97, Revision 2 (including approved deviations) be completed before startup following the first refueling outage of Unit 1.

7.7 Control Systems

7.7.2 Specific Findings

7.7.2.1 Effects of Control System Failures (LRG-II Generic Issues 5-ICSB and 7-ICSB)

In Section 7.7.2.1 of the SER, the staff reported that the applicant had initiated a detailed study to determine what, if any, design or procedural changes are necessary to ensure that the effects of failures of any power sources, sensors, or sensor impulse lines (which are shared by two or more control systems) will not result in consequences outside the bounds of FSAR Chapter 15 analyses or beyond the capability of operator or safety systems. The staff considered completion of this detailed study to be an outstanding unresolved issue.

In Section 7.7.2.1 of SSER No. 4, the staff reported that, in a letter dated March 14, 1983, the applicant had submitted a detailed analysis that responded to the staff's concerns regarding the effects of failures of power sources, sensors, or sensor impulse lines shared by control systems. This analysis identified the systems and their controls that could affect plant safety because of their capacity in the plant design. Failure modes were postulated for the various

various power sources, the ensuing plant responses were calculated, and an enveloping FSAR Chapter 15 analysis was identified for the postulated failure.

The staff reviewed the basis of the applicant's detailed study and concluded, with reasonable assurance, that the consequences of shared-power source failures with the control systems are bounded by the analysis documented in FSAR Chapter 15. However, the applicant had not provided sufficient information regarding failures of shared sensors or sensor impulse lines. As an open item, the applicant was required to submit additional information regarding multiple control system failures caused by a failure of a shared sensor or sensor impulse line.

As a further matter, this issue pertains to IE Bulletin 79-22, which states in part that if non-safety grade or control equipment were subjected to the adverse environment of a high-energy line break, it could impact the adequacy of protective functions in mitigating the consequences of the high-energy line break. The applicant was requested to review the Perry design to determine whether the harsh environment associated with high-energy line breaks might cause control system malfunctions, resulting in consequences more severe than those analyzed in the FSAR, or beyond the capability of operators or safety systems. The applicant also addressed this aspect in the March 14, 1983, letter, providing a summary of the results of his review. The summary identified (1) the function required to prevent transients, (2) all major system components that could significantly impact those functions, and (3) the control and instrumentation dependencies that could cause the components or systems to interact, resulting in transients not previously analyzed. In addition, several adverse conditions were postulated with the resulting failures of the items identified above. For all of the adverse conditions postulated, the applicant stated that the results of the analysis showed that multiple control system malfunctions resulting from harsh environments caused by high-energy line breaks do not result in consequences more severe than those analyzed in FSAR Chapter 15, or beyond the capability of operators or safety systems.

On the basis of its review of the information provided by the applicant relative to IE Bulletin 79-22, the staff finds that subjecting the instrumentation and control systems to the effects of high-energy line breaks in the immediate vicinity of system components would be unlikely to result in consequences more severe that those previously analyzed. However, the staff's review of the effects of high-energy line breaks cannot be fully completed until the following additional information is provided by the applicant:

(1) Verification that a single active failure in the safety systems used to mitigate the consequences of high-energy line breaks was assumed in the analysis performed.

(2) A description of the harsh environments was assumed in the analysis performed, including a discussion of the effects of pressure, temperature, and humidity in addition to pipe whip and jet impingement.

The applicant was advised of the additional information required to fully resolve this issue in a letter dated June 22, 1983.

By letter dated September 6, 1983, the applicant provided the results of his review of control system failures pertaining to sensors serving multiple control systems and sensor impulse lines feeding multiple sensors. This review concluded that no single shared sensor failure or single impulse line failure feeding multiple sensors would cause the transient analysis contained in FSAR Section 15 to be exceeded, and, therefore, is encompassed within the bounds of that analysis. From its review of the applicant's findings, the staff concludes that its concerns regarding multiple control system failures due to failure of common sensors or sensor (instrument) lines in the Perry control system design is no longer at issue. However, this conclusion only resolves SER Outstanding Issue (14) as it pertains to LRG-II Generic Issue 5-ICSB. The portion of SER Outstanding Issue (14) with respect to LRG-II Generic Issue 7-ICSB (the effects of high-energy line breaks on control systems) has not yet been adequately responded to by the applicant to consider SER Outstanding Issue (14) to be fully resolved. To fully resolve this issue, the applicant must provide the following additional information pertaining to LRG-II Generic Issue 7-ICSB:

(1) An identification of the location (elevation/areas) that contain high-energy piping systems, and in which components for nonsafety-related control systems are located, for the adverse conditions alluded to by the applicant in his letter dated March 14, 1983.

(2) A detailed analysis for the "turbine trip without bypass event" (FSAR Section 15.2.3) in conjunction with a high-energy line break that causes a loss of feedwater heating (and subsequent increase in reactor power level). Without operator action, the staff is concerned that this event could lead to a higher reactor power level than previously analyzed in the FSAR.

(3) The results of a zone analysis and a plant walkdown. If a zone analysis is not used, the applicant must describe the procedure by which the locations of nonsafety-related control system components affected by high-energy line breaks were determined.

(4) Verification that no credit was taken in the analysis for nonsafety-related equipment (e.g., feedwater trip on Level 8) to mitigate the effects of high-energy line breaks and consequential control system failures.

(5) Verification that the consequences of the worst-case event combination considered in the applicant's analysis are bounded by a small fraction (<10%) of 10 CFR 100 guideline doses.

The applicant was advised of the need for this additional information by letter dated January 10, 1984.

By letter dated October 2, 1984, the applicant provided the necessary information as noted above; that is, the applicant (1) verified that a single active failure in the safety systems used to mitigate the consequences of a high-energy line break was assumed in the applicant's analysis; (2) identified the locations (elevation/areas) that contain high-energy piping systems, and in which components for nonsafety-related control systems are located, for the adverse conditions cited; (3) described the harsh environments assumed in the analysis and included a discussion of the effects of pressure, temperature, and humidity in addition to pipe whip and jet impingement; (4) provided the results of a zone analysis and a plant walkdown: (5) verified that no credit was taken in the analysis for non-safety-related equipment (e.g., feedwater trip on

Level 8 to mitigate the effects of high-energy line breaks and consequential control system failures; and (6) verified that the consequences of the worst-case event conbination considered in the applicant's analysis are bounded by a small fraction (10%) of 10 CFR 100 guideline doses.

The staff has reviewed the additional information furnished in the October 2, 1984, letter, and the relevant information provided in the FSAR and concludes that the applicant has satisfactorily responded to all of its concerns relating to the high-energy line break concern. On the basis of this conclusion and the conclusions contained in the applicant's study (which indicates that the radiological consequences of the worst-case event combinations are bounded by the radiological consequences currently provided for each Chapter 15 event; that is, dose consequences will not exceed 10% of 10 CFR 100 criteria), the staff finds Outstanding Issue (14), listed in Section 1.9 of the SER, has been fully resolved.

8 ELECTRIC POWER SYSTEMS

8.3 Onsite Emergency Power Systems

8.3.1 Alternating Current Power Systems

In Section 8.3.1 of SSER No. 4, the staff reported that it found that the diesel generator test commitments conformed with its positions and were deemed acceptable. In a letter dated February 23, 1984, the applicant was advised that, in view of the concerns raised on the reliability of the Transamerica DeLaval, Inc. (TDI), diesel generators and the addition of this outstanding issue in SSER No. 4, the staff's position on the diesel generator testing program for Perry may change, and that the extent to which the test program will change would be subject to the staff's review of the TDI Diesel Generator Group recommendations for resolving that outstanding issue. Therefore, the staff's acceptance of the test program was accordingly rescinded.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage Assembly

9.1.5 Overhead Heavy-Load-Handling System

As a result of the NRC's investigations under Generic Task A-36, "Control of Heavy Loads Near Spent Fuel," NUREG-0612 was developed. Following the issuance of NUREG-0612, entitled "Control of Heavy Loads at Nuclear Power Plants," an NRC generic letter dated December 22, 1980, was sent to all operating plant licensees, applicants for operating licenses, and holders of construction permits requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. In accordance with that generic letter, the applicant was required to review provisions for the handling and control of heavy loads in the Perry plant to determine the extent to which the guidelines of NUREG-0612 were satisfied, and where not satisfied, to commit to mutually agreeable changes and modifications to fully satisfy those guidelines. By letters dated June 14, 1981, January 7, 1982, September 15, 1982, and November 8, 1982, the applicant provided responses to the NRC generic letter.

In Section 9.1.5 of SSER No. 2, the staff reported on its review of the applicant's submittals, indicating that the review was not complete, and that when completed, its findings would be reported in a future SER supplement. Pending completion of its review, the staff imposed a condition regarding Perry's compliance with the guidelines of NUREG-0612 (Phase I, addressing Section 5.1.1 of Nureg-0612; Phase II, addressing Sections 5.1.2 through 5.1.6 of NUREG-0612). This was identified as License Condition (18) in Section 1.11 of the SER.

The staff and its consultant, Idaho National Engineering Laboratory (INEL), have since completed their review of the applicant's submittals for Perry Units 1 and 2. As a result of this review, INEL issued a technical evaluation report (TER), which is being added to the SER as Appendix K by this supplement.

The staff has reviewed the TER and concurs with the consultant's findings therein that the guidelines of NUREG-0612, Section 5.1.1, have been acceptably satisfied by the applicant. The staff has also determined that the applicant need not take any further action regarding Sections 5.1.2 through 5.1.6 of NUREG-0612 at this time. Therefore, SER License Condition (18) is no longer required and is accordingly being deleted in this supplement.

10 STEAM AND POWER CONVERSION SYSTEM

10.3 Main Steam Supply System

10.3.4 Steam Erosion Effect on Valves

In conjunction with the evaluation findings relative to steam erosion effects on pipe breaks, discussed in Section 3.6.1 of this supplement, the staff has considered the possibility that steam erosion will cause valve seat damage that results in high valve leakage rates in the main steam isolation valves (MSIVs). However, the staff has found no evidence that this damage is likely to occur. The Perry MSIV design includes valve seats that are coated (hard-faced) with an extremely hard cobalt derivative to prevent wear of the valve seating surfaces. In addition, a leakage control system is provided to control the effects of leakage from the MSIVs. The MSIV leakage rates are periodically measured and compared with the allowable leakage rates specified in the Perry Technical Specifications. Should this allowable leakage rate be exceeded, the plant is required to be shut down until the cause of the problem is determined and corrected.

Therefore, on the basis of the above and the findings discussed in Section 3.6.1 of this supplement, the staff concludes that steam erosion will not cause valve leakage rates that exceed allowable limits and the leakage control system will contain the effects of valve leakage in conformance with the guidance set forth in SRP Section 6.7 (NUREG-0800).

13 CONDUCT OF OPERATIONS

13.3 Emergency Plans

13.3.1 Introduction

SSER No. 4 provided the staff's review and evaluation of the applicant's radiological emergency response plan (Plan) through Revision 2, dated August 5, 1983, submitted by letter dated September 16, 1983. In SSER No. 4, Section 13.3.3, the staff concluded that, on satisfactory correction of those items requiring resolution and commitment as identified in Section 13.3.2 of SSER No. 4, the Perry Emergency Plan will provide an adequate planning basis for an acceptable state of emergency preparedness.

After SSER No. 4 was issued, the applicant continued to upgrade emergency response planning and issued Revision 3 to the Plan, dated April 23, 1984, by letter dated April 28, 1984. In addition, submittals dated August 20, 1984, October 29, 1984, and January 16, 1985, provided additional clarification and commitments.

The staff has completed its review and evaluation of the adequacy of the applicant's revised Emergency Plan (through Revision 3) and the applicant's information and commitments provided by correspondence dated April 28, 1984, August 20, 1984, October 29, 1984, and January 16, 1985. The results of this evaluation are given in Section 13.3.2 below.

The acceptance criteria used as the basis for the staff's review are in SRP Section 13.3 (NUREG-0800), and include 10 CFR 50.47(b); Appendix E to 10 CFR 50; NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants"; and IE Information Notice 83-28, "Criteria For Protective Action Recommendations for General Emergencies." The guidance criteria of NUREG-0654 have been endorsed in RG 1.101, Revision 2, dated October 1981, and thus have the same status as a regulatory guide.

The staff's findings are discussed under the headings and numbering system used in SSER No. 4. Section 13.3.3 provides the Federal Emergency Management Agency's interim findings on the adequacy of offsite plans, and Section 13.3.4 provides the staff's conclusions.

An exercise of the Perry Emergency Plan was conducted at the Perry Nuclear Power Plant on November 28, 1984, and documented in IE Inspection Report Nos. 50-440/84-24 and 50-441/84-22, dated December 19, 1984. The exercise tested the applicant's onsite and offsite emergency support organizations' capabilities to respond to a simulated accident scenario resulting in a major radioactive release. The exercise was integrated with a test of the State of Ohio, State of Pennsylvania, Lake County, Geauga County, and Ashtabula County Emergency Plans. This was a full-participation exercise for these counties and the State of Ohio. Pennsylvania participation was limited to the ingestion pathway.

Although exercise weaknesses that require corrective action were identified by the NRC staff, the exercise report concludes: "The applicant's response was generally coordinated, orderly and timely. Had these events been real, actions taken by the applicant would have been sufficient to permit State and local authorities to take appropriate actions to protect the public's health and safety."

13.3.2 Evaluation of the Emergency (Onsite) Plan

13.3.2.2 Onsite Emergency Organization

Unresolved Item (1)

The applicant's concept of augmented shift staffing, with respect to health physics capabilities, is not consistent with the guidance found in Table 2 of Supplement 1 to NUREG-0737. The applicant should either change his concept to be consistent with the guidance or provide alternative means for achieving the goals set forth in the guidance.

Evaluation

In a letter dated April 28, 1984, the applicant explained that, in keeping with the policy of assigning responsibilities during an emergency to reflect normal functions, the Chemistry Unit Supervisor will perform emergency dose assessment. Accordingly, the 60-min column of Table 5-1 indicates that the Chemistry Unit Supervisor (previously the Plant Health Physicist) is responsible for offsite dose assessment at the emergency operations facility (EOF). The staff finds this portion of the applicant's Plan adequate.

The unresolved item pertaining to augmented staffing by radiation monitoring teams is addressed in Section 13.3.2.9 of this supplement.

Unresolved Item (2)

Additional information should be provided that describes in more detail the transfer process of responsibilities for the position of Emergency Coordinator. In particular, the Plan should describe the briefing information necessary for the Vice-President, Nuclear Group (or alternative), to assume the position of Emergency Coordinator.

Evaluation

Section 6.1.3 of the Plan has been revised to include the briefing information that will be provided to the Vice-President, Nuclear Group, before his assumption of the position of Emergency Coordinator. This information will include plant status, dose projections, offsite protective actions and recommendations, notification status, and any other information pertinent to the emergency response given by the individual currently acting as Emergency Coordinator. The staff finds this portion of the applicant's Plan adequate.

Unresolved Item (3)

The Plan should make clear that the nondelegable duties of notification and protective action recommendations to offsite authorities are the responsibility of each individual who assumes the role of Emergency Coordinator.

Evaluation

The Plan has been revised to show that the Shift Supervisor (Section 5.2.2.1), Operations Manager (Section 5.2.2.3), and the Vice-President, Nuclear Group (Section 5.2.2.4), upon their assuming the position of Emergency Coordinator, have the nondelegable responsibilities of the decision to notify and make protective action recommendations to offsite authorities. The staff finds this portion of the applicant's Plan adequate.

Unresolved Item (4)

Clarification that the individual in the position of Emergency Coordinator maintains unilateral control of the overall emergency response should be provided in the Plan.

Evaluation

The Plan has been revised to specify that any individual fulfilling the position of Emergency Coordinator has the responsibility and the authority to direct any and all phases of emergency response. The staff finds this portion of the applicant's Plan adequate.

13.3.2.3 Emergency Response Support and Resources

Unresolved Item (1)

The Plan should indicate the expected arrival times of Federal assistance at the site.

Evaluation

The Plan has been revised to specify the capability of the Department of Energy (DOE), Chicago Operations Office, and the arrival times for various DOE Region V radiological assistance program teams. The staff finds this portion of the applicant's Plan adequate.

Unresolved Items (2) and (3)

Clarification of the radiation detection capabilities of the postaccident sampling system (PASS) should be provided in the Plan. In addition, more information concerning the capabilities of the laboratory equipment at the technical support center (TSC) and EOF should be provided.

Provisions consistent with NUREG-0737, Item II.B.3, should be made for backup laboratory facilities capable of analyzing high-level samples.

Evaluation

Section 7.3.9 of the Plan has been revised to clarify the use and capabilities of the PASS. The Plan specifies that the PASS provides samples for onsite and offsite laboratories for wet chemistry and isotopic analysis. Further, the Plan states that the shielding study conducted in response to NUREG-0737, TMI Action Plan Item II.B.2, shows that the normal counting and laboratory facilities will be habitable under postaccident conditions. Therefore, the normal counting room will be used to perform isotopic analysis of PASS samples, and the normal laboratories will be used to provide backup liquid analysis capabilities for those analyses done in line. The capabilities of the normal counting room and laboratories (including a high-level laboratory) are addressed in Section 12.5.2 of the SER. Revision 3 to the Plan describes the capability of the equipment at the backup counting room in the TSC.

In addition, in a letter dated August 20, 1984, the applicant specified that the PASS description and radiation detection capabilities and provisions for backup laboratory facilities for analyzing high-level samples are detailed in September 16, 1983, and October 14, 1983, submittals on TMI Action Plan Item II.B.3. The staff's review and evaluation of the PASS is contained in Section 9.3.2 of SSER No. 4, which includes, in part, a review and evaluation of the PASS against Item II.B.3, Criterion 8 (capability for backup sampling), Criterion 9 (radiological and chemical capabilities), and Criterion 10 (accuracy, range, and sensitivity of instruments and procedures). The staff concluded that the PASS design meets the applicable criteria of NUREG-0737, Item II.B.3, and is thus considered to be acceptable. On the basis of the staff's review of Revision 3 to the Plan and the conclusions related to the PASS design in Section 9.3.2 of SSER No. 4, the staff finds this portion of the applicant's Plan adequate.

In a letter dated October 29, 1984, the applicant committed to describe in the Plan the capability of the portable counting equipment at the EOF to be used for analyzing samples obtained by the radiation monitoring teams. This matter is confirmatory, pending a revision to the Plan.

13.3.2.4 Emergency Classification System

Unresolved Item (1)

The Plan should incorporate into the Perry emergency classification scheme the comments forwarded to the applicant as noted in NRC letter dated January 11, 1984.

Evaluation

The applicant has modified the emergency classification scheme (Table 4-1) of the Plan in light of the staff's comments in the letter dated January 11, 1984. With regard to the staff's comment on Site Area Emergency initiation condition 15b (flood, low water), in a letter dated April 18, 1984, the applicant explained that the emergency action level (EAL) for low water was not incorporated because the location of the Perry site and the design of the lake intake structures preclude the sensitivity to low lake level that may be true at other coastal sites. The Plan contains EALs for low water associated with Notification of Unusual Event (lake level is below 565 ft, U.S. Geological Survey (USGS)) and Alert (lake level is below 560 ft, USGS). In addition, the Plan contains an EAL for the Site Area Emergency level if the shift supervisor determines that "other plant conditions exist that warrant activation of emergency response facilities or precautionary public notification." With regard to the staff's comment on Site Area Emergency initiating condition 1 (known LOCA), the Plan was revised, and on August 20, 1984, the applicant provided his rationale for the use of AND logic rather than OR logic in the EAL dealing with an initiating condition of known loss of reactor coolant greater than makeup pump capacity. The above two EALs have been adequately corrected by the applicant, and the staff finds this portion of the applicant's Plan adequate.

With regard to Site Area Emergency initiating condition 2 (degraded core), Revision 3 to the Plan included a modification to this EAL, and on August 20, 1984, the applicant proposed a further change to EAL 2 and committed to include this change in Revision 4 to the Plan. The proposed change appears to be acceptable. This matter is confirmatory, pending the revision to the Plan.

Unresolved Item (2)

The Plan should indicate, consistent with 10 CFR 50, Appendix E.IV.B, that the EALs are agreed upon between the applicant and the State/local authorities and that they will be reviewed annually.

Evaluation

Revision 3 to the Plan addressed the matter of review of EALs with State and local officials, and on August 20, 1984, the applicant submitted a proposed change to the Plan that includes the aspect that EALs will be initially agreed on by State and local officials, as well as an annual review. The staff finds the applicant's response acceptable. This matter is confirmatory, pending the change to Section 8.2 of the Plan.

Unresolved Item (3)

Protective action recommendations based on the potential for release of radioactive materials should be more thoroughly discussed in the Plan. Specifically, the Plan should discuss: (a) other nonradiological means of core status determination; (b) the relationship between gap activity and percentage of failed fuel as it relates to protective action recommendations; and (c) how the core status graph found in Figure 4-1 of the Plan will be used in the emergency action level scheme.

Evaluation

As a result of further review of Revisions 2 and 3 to the Plan, the staff concludes that Item (a) above is in fact satisfactorily addressed in the current Plan. This matter was the subject of a staff letter to the applicant dated February 29, 1984. With regard to Items (b) and (c), the Plan has been revised to show how the relationship between the radiation reading at the containment monitor and percent failed fuel (Figure 4-1) will be utilized in the EAL scheme. Sections 4.1.4 and 6.4.3 of the Plan provide for immediate protective actions out to 5 mi to be recommended to offsite authorities on the basis of containment monitor readings as applied to Figure 4-1 for General Emergency conditions.

However, the Plan does not meet the requirements of 10 CFR 50.47(b)(10) with regard to a range of protective actions for the plume emergency planning zone (EPZ) for emergency workers and the public. The applicant proposed further changes to the Plan on October 29, 1984, and January 16, 1985, with regard to establishing protective action recommendations for the entire plume EPZ during general emergencies. The proposed changes to the Plan indicate that assessment activities based on actual or potential radioactive releases, as described in the emergency planning instructions (EPIs), will determine if additional protective actions (i.e., beyond 5 mi) should be recommended. Draft procedure EPI-B8 provides instructions on defining the effects of the release and, depending on meteorological conditions, population distribution, and condition of roads and major traffic ways, expanding existing protective action recommendations.

With regard to Item (b), the staff finds this portion of the applicant's Plan adequate. With regard to Item (c), this matter is confirmatory pending the revision to the Plan.

13.3.2.5 Notification Methods and Procedures

Unresolved Item (1)

The applicant should develop methods agreed upon by the applicant and the county authorities that will allow for prompt notification of the public in a rapidly escalating emergency. A description of these methods should appear in the Plan.

Evaluation

Section 6.4.2 of the Plan has been revised to include the activation of the prompt alerting system and placing a message implementing the Perry recommendation on the emergency broadcast system (EBS) by the county dispatcher in the event higher authority cannot be contacted. Further discussion of this matter, provided by the the applicant on August 20, 1984, specifies that current drafts of the standard operating procedure (SOP) for the county emergency communications center (ECC) provide instructions to the ECC Watch Commander in the event of a rapidly escalating emergency. If one of the county commissioners or the Disaster Services Agency Director cannot be reached within 12 min of the time of notification, the ECC Watch Commander will act appropriately in accordance with his instructions as outlined in the SOP with regard to activation of the sirens and EBS messages. The staff finds this portion of the applicant's Plan adequate.

The staff will require the applicant to demonstrate that the prompt alerting system, as described in Section 7.2.5 of the Plan, is installed and operational before fuel loading. The determination of conformance of the overall alert and notification system with the guidance of Appendix 3 to NUREG-0654 will be provided at a later date by the Federal Emergency Management Agency (FEMA) in the course of its review and administrative approval of offsite emergency preparedness under 44 CFR 350 of FEMA's rules.

Unresolved Item (2)

The Plan should address the periodic testing of sirens within the plume exposure EPZ. Appendix 3 of NUREG-0654 should be consulted for guidance.

Evaluation

Revision 3 to the Plan specifies that testing of the prompt alerting system will be conducted in accordance with NUREG-0654, Revision 1, Appendix 3. The staff finds this portion of the applicant's Plan adequate.

13.3.2.6 Emergency Communications

Unresolved Item (1)

The applicant should provide for backup communications capabilities between the site and all State/local governmental authorities with primary responsibilities during an emergency.

Evaluation

In letters dated August 20, 1984, and October 29, 1984, the applicant committed to revise Figure 7-6 and Section 7.2.2 of the Plan to clarify the backup communication links between the site and State/local authorities. This matter is confirmatory, pending revision of the Plan.

Unresolved Item (2)

The applicant should provide for a coordinated communication link for fixed and mobile medical support facilities.

Evaluation

In a letter dated August 20, 1984, the applicant committed to revise Section 7.2.2.8 of the Plan to describe the communications between the site, Perry Township Fire Department, Lake County Memorial Hospital, and the fire department's ambulance. This matter is confirmatory, pending the revision of Section 7.2.2.8 of the Plan.

13.3.2.7 Public Information

Unresolved Item (1)

The applicant should provide finalized emergency information brochures and other emergency information materials to the public prior to fuel load.

Evaluation

In Revision 3 to the Plan, the applicant commits to initially distribute an emergency information booklet to households and places of business within the plume exposure pathway EPZ before the issuance of an operating license. The staff finds this portion of the applicant's Plan adequate.

Unresolved Item (2)

The Plan should specify the location of the joint public information center (JPIC).

Evaluation

Section 7.1.4 of the Plan (Rev. 3) specifies the location of the JPIC as Lakeland Community College, Kirtland, Ohio, about 12 mi from the site. The staff finds this portion of the applicant's Plan adequate.

13.3.2.8 Emergency Facilities and Equipment

Unresolved Item (1)

The Plan should specify the time required, after activation, to bring the TSC and EOF to functional readiness.

Evaluation

Revision 3 to the Plan specifies that the TSC and EOF are expected to be fully functional within 15 min after the arrival of their respective staffs. The staff finds this portion of the applicant's Plan adequate.

Unresolved Item (2)

The Plan should indicate that emergency equipment and supplies will be available in the control room. Further, calibration frequency for emergency equipment should be specified, and should indicate that instruments removed from service will be replaced with a comparable instrument.

Evaluation

As a result of further review of Revision 2 to the Plan, the staff concludes that the calibration frequency for emergency equipment was adequately covered. This matter was addressed by the staff in a letter dated February 29, 1984, and is considered closed. In a letter dated August 20, 1984, the applicant committed to revise the Plan to indicate that emergency equipment, including self-contained breathing apparatus, will be available to the control room for emergency use. This matter is confirmatory, pending the revision to the Plan.

Unresolved Item (3)

Consistent with Supplement 1 of NUREG-0737, the Plan should describe the capability to obtain 24 hour-per-day regional (up to 10 mi) weather information.

Evaluation

In a letter dated August 20, 1984, the applicant described his intent to obtain a letter of agreement from the Cleveland-Hopkins branch of the National Weather Service to provide backup meteorology for the site. The Plan will be revised and a copy of the letter will be appended to the Plan. This matter is confirmatory, pending the revision to the Plan.

Unresolved Item (4)

The applicant should provide a commitment that the permanent emergency response facilities and equipment will be operational prior to fuel loading, or that adequate interim facilities and capabilities will be in place.

Evaluation

The applicant in his response (April 15, 1983) to Generic Letter 82-33 commits to have the technical support center, emergency operations facility, and operations support center (OSC) fully functional by fuel load of Unit 1. The applicant's current construction completion date is December 1985. The staff finds this portion of the applicant's Plan adequate.

13.3.2.9 Accident Assessment

Unresolved Item

Additional information concerning the staffing and capabilities of the radiation monitoring teams (RMTs) should be provided in the Plan. In particular, the Plan should provide information about the availability of transportation. Also, the level of staffing for RMTs, starting with the Alert classification stage, should be consistent with the guidance found in Table 2 of Supplement 1 to NUREG-0737.

Evaluation

Section 5.2.2.4 of the Plan (Rev. 3) addresses the availability of transportation for RMTs and generally describes the equipment for offsite radiological measurements and communications. The staff finds this portion of the applicant's Plan adequate.

Table 5-1 of Revision 3 to the Plan is not consistent with Table 2 of Supplement 1 to NUREG-0737, with regard to augmenting the shift health physics technician. However, the applicant in his submittal of October 29, 1984, commits to provide three RMTs, each composed of one instrument and control technician and one radiation protection section technician. Two teams will be available at 30 min to be utilized in all areas of the EPZ (i.e., off site and on site (out of plant) to effectively monitor radioactive releases). An appropriate Plan change will be made. This matter is confirmatory, pending revision to the Plan.

13.3.2.10 Protective Response

Unresolved Item (1)

The Plan should describe in more detail the provisions for evacuating visitor and/or contractor personnel. Evacuation routes and assembly area locations for onsite personnel should be illustrated in the Plan. In addition, the Plan should describe in more detail the measures to be taken to compensate for the effects, on the evacuation of onsite personnel, of inclement weather, high traffic density, and specific radiological conditions. Further, the Plan should specify the time required to warn onsite individuals that an emergency situation exists.

Evaluation

Revision 3 to the Plan addresses 30-min accountability, evacuation of nonessential personnel, evacuation routes, and alternate offsite assembly areas in the event of adverse weather, radiological, or traffic conditions. Figures 6-7 A, B, and C illustrate evacuation routes and assembly areas. Section 7.2.1 of the Plan describes a plant public address system and an exclusion area paging system used to broadcast a prerecorded message that is audible over the entire exclusion area. Emergency Planning Instructions EPI-B5, "Personnel Accountability," and EPI-B6, "Evacuation," provide instructions for immediately notifying personnel within the exclusion area. The staff finds this portion of the applicant's Plan adequate.

In a letter dated October 29, 1984, the applicant provided additional information on the time required to warn onsite personnel, which will be included in the next revision to the Plan. This matter is confirmatory, pending the revision to the Plan.

Unresolved Item (2)

Personnel monitoring methods should be described in the Plan, and decontamination supplies should be described with their location. In addition, the Plan should indicate that personnel accountability will be accomplished within 30 min.

Evaluation

Revision 3 of the Plan addresses the monitoring of personnel who are evacuated from the site to assembly areas, as well as those essential personnel reporting to the EOF. Emergency Plan instructions specify that normal health physics practices should be observed regarding contamination monitoring of personnel leaving the protected area. The Plan also describes decontamination supplies to be located in the EOF and OSC decontamination rooms. The Plan specifies that accountability will be performed of all personnel on site during Site Area or General Emergency level events. Accountability of personnel who do not perform an emergency response function (nonessential personnel) will be accomplished within 30 min. For those essential personnel within the site boundary (protected area), the fire/security computer will provide a list of personnel that will be compared with those who have reported to their assigned emergency response facility. The staff finds this portion of the applicant's Plan adequate.

Unresolved Item (3)

The evacuation time estimate study should address the effects of adverse weather (i.e., a thunderstorm) on a summer Sunday evacuation. The Plan should indicate that the evacuation time estimates have been reviewed by the appropriate State and local officials.

Evaluation

An updated evacuation time estimate (ETE) study that addresses adverse weather conditions is included as Appendix D to the Plan. The ETE study, dated March

1984, was reviewed against the guidance of NUREG-0654, Appendix 4, and the staff finds this portion of the applicant's Plan adequate.

In a letter dated April 28, 1984, the applicant specified that the ETE study was being reviewed with State and local officials and a letter of concurrence is expected from each. This matter is confirmatory, pending the applicant's submittal of copies of the concurrence letters for the staff's review.

13.3.2.11 Radiological Exposure Control

Unresolved Item (1)

The Plan should indicate that each emergency worker will be issued both a self-reading and a permanent recording dosimeter. Further, the Plan should indicate that the emergency personnel dosimetry program includes the capability for 24-hour-per-day determination of doses.

Evaluation

Revision 3 to the Plan specifies the use of self-reading dosimeters and thermo-luminescent dosimeters, as well as the capability to determine an individual's exposure on a 24-hour-per-day basis. The staff finds this portion of the applicant's Plan adequate.

Unresolved Item (2)

Contamination action levels for personnel and equipment should be specified in the Plan. Additionally, the Plan should indicate that decontamination supplies include materials capable of radioiodine skin decontamination.

Evaluation

Revision 3 to the Plan specifies contamination action levels for personnel, equipment, and areas. The Plan also includes a list of decontaminators that would be used for removing difficult-to-remove contamination. The staff finds this portion of the applicant's Plan adequate.

13.3.2.12 Medical and Public Health Support

Unresolved Item (1)

A letter of agreement with Northwestern Memorial Hospital should be included in the Plan.

Evaluation

In a letter dated April 28, 1984, the applicant explained that the agreement with Northwestern Memorial Hospital is made by Radiation Management Corporation and not by CEI. Section 5.3.3.2 of the Plan has been revised to clarify this matter. The staff finds this portion of the applicant's Plan adequate.

Unresolved Item (2)

A more detailed discussion of the availability of and training for personnel designated to provide first aid assistance should be included in the Plan. In particular, the Plan should indicate that these personnel will, at a minimum, receive first aid instruction at least equivalent to the Red Cross Multi-Media training, and that they will be available on a 24-hour-per-day basis.

Evaluation

Revision 3 to the Plan describes first aid members (security force) available on a 24-hour-per-day basis. For injuries inside the radiation-controlled area, the onshift health physics technician will respond to assist. The Plan specifies that first aid members receive Multi-Media Red Cross training or equivalent. The staff finds this portion of the applicant's Plan adequate.

13.3.2.14 Exercises and Drills

Unresolved Item (1)

The Plan should be revised to accurately and clearly reflect the requirements of 10 CFR 50, Appendix IV.F, and other applicable positions adopted by the Commission with respect to emergency exercises and drill frequency.

Evaluation

Section 8.5.4 of the Plan has been revised to clarify the drill and exercise program proposed by the applicant. The proposed additional changes to Sections 8.5.4.1.2 and 8.5.4.1.3 in the applicant's letter dated August 20, 1984, appear satisfactory. These proposed changes clarify the frequency of participation by State and local agencies. In a letter dated October 29, 1984, the applicant proposed a further revision to the Plan to clarify the annual frequency of the exercises. This matter is confirmatory, pending the revision to the Plan.

Unresolved Item (2)

The Plan should indicate that the exercise scenario will be varied over a 5-year period to ensure that all major portions of the emergency plan are tested. Further, the Plan should specify that each exercise will include testing of the public notification system and that exercises will be conducted under various weather conditions.

Evaluation

Revision 3 to the Plan specifies that the scenario for the annual emergency plan exercise will be varied from year to year to ensure that all major portions of the Plan are tested over a 5-year period. The Plan includes testing of the prompt alerting system during each exercise and provides for conducting the annual exercise under varying weather conditions. The staff finds this portion of the applicant's Plan adequate.

Unresolved Item (3)

The Plan should indicate that part of each communication drill will involve evaluating the aspect of understanding the contents of the message.

Evaluation

In a letter dated August 20, 1984, the applicant committed to revise Section 8.5.4.3 of the Plan so that the communications tests described in the Plan will involve evaluating the understanding of the contents of the message, including a check of hardware, voice quality, and message clarity. This matter is confirmatory, pending revision to the Plan.

13.3.2.15 Radiological Emergency Response Training

Unresolved Item

The Plan should indicate that initial and annual retraining of emergency personnel will be provided.

Evaluation

Revision 3 to the Plan specifies retraining on an annual basis for emergency organization personnel who receive specialized training. The staff finds this portion of the applicant's Plan adequate.

13.3.2.16 Responsibility for the Planning Effort: Development, Periodic Review, and Distribution of Emergency Plans

Unresolved Item (1)

The Plan should specify that the Nuclear Safety Review Committee (NSRC) has no direct responsibility for emergency preparedness planning at Perry. In addition, the scope of the NSRC review should be described in more detail, and the Plan should indicate that administrative means are in place to ensure correction of deficient areas and incorporation of the results into the emergency preparedness program.

Evaluation

Revision 3 to the Plan specifies that the NSRC has no direct responsibility for emergency preparedness planning at Perry. The Plan describes the scope of the annual review whose purpose it is to verify compliance with the CEI Quality Assurance Program, internal rules and procedures, Federal regulations, and operating license conditions. The results of the annual review are documented in a report to the Vice President-Nuclear Group by the NSRC. The Plan and its implementing instructions are controlled and revised in accordance with Perry administrative procedures. The staff finds this portion of the applicant's Plan adequate.

In a letter dated October 29, 1984, the applicant proposed a Plan change that addresses the aspect of implementing management controls for evaluation and correction of review findings and incorporation of corrections into the emergency preparedness program. This matter is confirmatory, pending the revision to the Plan.

Unresolved Item (2)

The administrative procedures used to revise and update the Plan and implementing procedures should be described in more detail. In particular, the Plan should indicate that revised pages are dated and marked to show where changes have been made.

Evaluation

Section 8.2 of the Plan has been revised to include information on review and approval of the Plan and implementing instructions as well as control and distribution in accordance with Perry administrative procedures. In a letter dated April 28, 1984, the applicant explained that the administrative procedures provide for pages to show a revision number and to indicate changes made with a side bar. The applicant's Revision 3 to the Plan adheres to these procedures. The staff finds this portion of the applicant's Plan adequate.

Unresolved Item (3)

Appendix F, the cross reference between NUREG-0654 elements and Plan sections, should be updated.

Evaluation

Revision 3 to the Plan provided an update on Appendix F to the Plan. The staff finds this portion of the applicant's Plan adequate.

13.3.3 Review of State and Local Plans by the Federal Emergency Management Agency

SSER No. 4 noted that the Federal Emergency Management Agency (FEMA) would review State and local emergency response plans and make a finding on the state of offsite planning and preparedness. FEMA's interim findings on planning are provided in Appendix L to this supplement. In the letter dated March 1, 1984, in Appendix L, FEMA stated: "Based on the Region V review of the Ohio State and Ashtabula, Geauga, and Lake Counties offsite radiological emergency preparedness plans, there is reasonable assurance that the plans are adequate and capable of being implemented in the event of an accident at the site."

A full-participation exercise to test the onsite and offsite plans was held on November 28, 1984. FEMA's findings on whether the exercise demonstrated that offsite emergency preparedness is adequate will be provided in a future supplement.

13.3.4 Conclusions

On the basis of the staff's review of the applicant's Plan, the staff concludes that, upon satisfactory completion of those items committed to by the applicant as identified in Section 13.3.2 of this supplement, the Perry radiological emergency plan will provide an adequate planning basis for an acceptable state of onsite emergency preparedness and will meet the requirements of 10 CFR 50 and Appendix E thereto. SER Outstanding Issue (19) is accordingly being changed to Confirmatory Issue (61) by this supplement. The staff will confirm that the applicant has complied with his commitments in a future supplement.

On the basis of a review of State and local plans, FEMA has reported that there is reasonable assurance that the plans are adequate and capable of being implemented. FEMA's exercise evaluation report has not yet been received. FEMA's findings on the adequacy of offsite emergency preparedness as demonstrated by the exercise will be provided in a future supplement.

13.6 Physical Security

In Section 13.6 of the SER, the staff proposed that a license condition be required for the applicant to fully implement and maintain in effect all provisions of the staff-approved physical security, guard training and qualification, and safeguards contingency plans, including amendments made pursuant to the authority of 10 CFR 50.54(p). Accordingly, this condition was listed as License Condition (13) in Section 1.11 of the SER. The staff's primary evaluation documented in the SER was predicated on the following security program plans filed with the NRC by the applicant for meeting the requirements of 10 CFR 73:

(1) Perry Nuclear Power Plant Physical Security Plan
(2) Security Force Training and Qualification Plan
(3) Safeguards Contingency Plan (Chapter 8 of the Security Plan)

This supplement documents the basic analysis that is available for public review and in a protected appendix, predicated on information provided by the applicant since the SER was issued in May 1982. On the basis of a review of these documents and visits to the plant site, the staff has concluded that the protection by the applicant meets the requirements of 10 CFR 73, and accordingly the protection planned will ensure that the health and safety of the public will not be endangered.

By letters dated July 18, 1984, and September 19, 1984, the applicant submitted two changes to the physical security plan which are still under staff review. These changes contain safeguards information and are, therefore, withheld from public disclosure in accordance with 10 CFR 73.21. The staff's findings on these changes will be addressed in a future SER supplement. The evaluation that follows may be revised as a result of this review.

13.6.1 Physical Security Organization

To satisfy the requirements of 10 CFR 73.55(b), the applicant has provided a physical security organization that includes a security shift supervisor who is on site at all times with the authority to direct the physical protection activities. To implement the commitments made in the physical security plan, training and qualification plan, and the safeguards contingency plan, written security procedures specifying the duties of the security organization members have been developed and are available for inspection. The training program and critical security tasks and duties for the security organization personnel are defined in the "Perry Security Force Training and Qualification Plan," which meets the requirements of 10 CFR 73, Appendix B, for the training, equipping, and requalification of the security organization members. The physical security plan and the training program provide commitments that preclude the assignment of any individual to a security-related duty or task before the individual is trained, equipped, and qualified to perform the assigned duty in accordance with the approved guard training and qualification plan.

13.6.2 Physical Barriers

To meet the requirements of 10 CFR 73.55(c), the applicant has provided a protected area barrier that meets the definition in 10 CFR 73.2(f)(1). An isolation zone, to permit observation of activities along the barrier, is provided as follows (except for the locations listed in the protected appendix): at least 20 ft inside the inner perimeter fence and 20 ft between the outer and inner perimeter fences.

The staff has reviewed those locations and determined that the security measures in place are satisfactory and continue to meet the requirements of 10 CFR 73.55(c).

Illumination of 0.2 ft-candles is maintained for the isolation zones, protected area barriers, and external portions of the protected area.

13.6.3 Identification of Vital Areas

The protected appendix contains a discussion of the applicant's vital area program and identifies those areas and items of equipment determined to be vital for protection purposes. Vital equipment is located within vital areas that are located within the protected area and that require passage through at least two barriers, as defined in 10 CFR 73.2(f)(1) and (2), to gain access to the vital equipment. Vital area barriers are separated from the protected area barrier.

The control room and central alarm station are provided with bullet-resistant walls, doors, ceilings, floors, and windows. On the basis of these findings and the analysis set forth in Paragraph C of the protected appendix, the staff has concluded that the applicant's program for identification and protection of vital equipment satisfies the regulatory intent. However, this program is subject to onsite validation by the staff in the future and to subsequent changes if found to be necessary.

13.6.4 Access Requirements

In accordance with 10 CFR 73.55(d), all points of personnel and vehicle access to the protected area are controlled. The individual responsible for controlling the final point of access into the protected area is located in a bullet-resistant structure. As part of the access control program, vehicles (except under emergency conditions), personnel, packages, and materials entering the protected area are searched for explosives, firearms, and incendiary devices by electronic search equipment and/or physical search.

Vehicles admitted to the protected area, except applicant-designated vehicles, are controlled by escorts. Applicant-designated vehicles are limited to onsite station functions and remain in the protected area except for operational maintenance, repair, security, and emergency purposes. Positive control over the vehicles is maintained by personnel authorized to use the vehicles or by the escort personnel.

A picture badge/key card system, utilizing encoded information, identifies individuals that are authorized unescorted access to protected and vital areas and is used to control access to these areas. Individuals not authorized unescorted access are issued non-picture badges that indicate an escort is required. Access authorizations are limited to those individuals who have a need for access to perform their duties.

Unoccupied vital areas are locked and alarmed. During periods of refueling or major maintenance, access to the reactor containments(s) is positively controlled by a member of the security organization to ensure that only authorized individuals and materials are permitted to enter. In addition, all doors and personnel/equipment hatches into the reactor containment(s) are locked and alarmed. Keys, locks, combinations, and related equipment are changed on an annual basis. In addition, when an individual's access authorization has been terminated because of lack of reliability or trustworthiness or because of poor work performance, the keys, locks, combinations, and related equipment to which that person had access are changed.

13.6.5 Detection Aids

To satisfy the requirements of 10 CFR 73.55(e), the applicant has installed intrusion detection systems at the protected area barrier, at entrances to vital areas, and at all emergency exits. Alarms from the intrusion detection system annunciate within the continuously manned central alarm station and a secondary alarm station located within the protected area. The central alarm station is located so that the interior of the station is not visible from outside the perimeter of the protected area. In addition, the central station is constructed so that walls, floors, ceilings, doors, and windows are bullet resistant. The alarm stations are located and designed in such a manner that a single act cannot interdict the capability of calling for assistance or responding to alarms. In the central alarm station, no other functions or duties that would interfere with its alarm response function are performed. The intrusion detection system transmission lines and associated alarm annunciation hardware are self-checking and tamper-indicating. Alarm annunciators indicate the type of alarm and its locaton when activated. An automatic indication of when the alarm system is on standby power is provided in the central alarm station.

13.6.6 Communications

As required in 10 CFR 73.55(f), the applicant has provided for the capability of continuous communications between the central and secondary alarm station operators, guards, watchmen, and armed response personnel through the use of a conventional telephone system and a security radio system. In addition, direct communication with the local law enforcement authorities is maintained through the use of a conventional telephone system and two-way FM radio links. All nonportable communication links, except the conventional telephone system, are provided with an uninterruptible emergency power source.

13.6.7 Test and Maintenance Requirements

To meet the requirements of 10 CFR 73.55(g), the applicant has established a program for the testing and maintenance of all intrusion alarms, emergency alarms, communication equipment, physical barriers, and other security-related devices and equipment. Equipment or devices that do not meet the design performance criteria or have failed to otherwise operate will be compensated for by appropriate compensatory measures as defined in the "Perry Nuclear Power Plant Security Plan" and in site procedures. The compensatory measures defined in these plans will ensure that the effectivenss of the security system is not reduced by failures or other contingencies affecting the operation of the security-related equipment or structures. Intrusion detection systems are tested for proper performance at the beginning and end of any period during which they are used for security. Such testing will be conducted at least once every 7 days.

Communication systems for onsite communications are tested at the beginning of each security shift. Offsite communications are tested at least once each day.

Audits of the security program are conducted once every 12 months by personnel independent of site security management and supervision. The audits, focusing on the effectivenss of the physical protection provided by the onsite security organization implementing the approved security program plans, include, but are not limited to, a review of the security procedures and practices, system testing and maintenance programs, and local law enforcement assistance agreements. A report is prepared documenting audit findings and recommendations and is submitted to the plant management.

13.6.8 Response Requirements

To meet the requirements of 10 CFR 73.55(h), the applicant has provided for armed responders immediately available for response duties on all shifts consistent with the requirements of the regulations. Considerations used in support of this number are attached in the protected appendix. In addition, liaison with local law enforcement authorities to provide additional response support in the event of security events has been established and documented.

The applicant's safeguards contingency plan for dealing with thefts, threats, and radiological sabotage events satisfies the requirements of 10 CFR 73, Appendix C. The plan identifies appropriate security events that could initiate a radiological sabotage event and identifies the applicant's preplanning, response resources, safequards contingency participants, and coordination activities for each identified event. Through this plan, on detection of abnormal presence or activities within the protected or vital areas, response activities using the available resources would be initiated. The response activities and objectives include the neutralization of the existing threat by requiring the response force members to interpose themselves between the adversary and the objective, instructions to use force commensurate with that used by the adversary, and authority to request sufficient assistance from the local law enforcement authorities to maintain control over the situation.

To assist in the assessment/response activities, a closed-circuit television system, providing the capability to observe the entire protected area perimeter, isolation zones, and a majority of the protected area, is provided to the security organization.

13.6.9 Employee Screening Program

To meet the requirements of 10 CFR 73.55(a) to protect against the design-basis threat as stated in 10 CFR 73.1(a)(1)(ii), the applicant has provided an employee screening program. Personnel who successfully complete the employee screening program or its equivalent may be granted unescorted access to protected and vital areas at the Perry site. All other personnel requiring access to the site are escorted by persons authorized and trained for escort duties and who have successfully completed the employee screening program. The employee screening program is based on accepted industry standards and includes a background investigation, a psychological evaluation, and a continuing observation program. In addition, the applicant may recognize the screening program of other nuclear utilities or contractors on the basis of a comparability review conducted by the Cleveland Electric Illuminating Company.

The plan also provides for a "grandfather clause" exclusion, which allows recognition of a certain period of trustworthy service with the utility or contractor as being equivalent to the overall employee screening program. The staff has reviewed the applicant's screening program against the accepted industry standards (American National Standards Institute Standard N18.17 1973) and has determined that the program is acceptable.

In view of the basic evaluation presented above, License Condition (13), listed in Section 1.11 of the SER, is being amended in this supplement to more specifically cite the condition that will be incorporated in the operating license to be issued for Perry Units 1 and 2.

15 TRANSIENT AND ACCIDENT ANALYSIS

15.4 Rod Withdrawal Events

15.4.2 Rod Withdrawal Error at Power

In a letter dated June 29, 1984 (D.L. Holtzscher to J. J. Stephens), the Licensing Review Group (LRG)-II, a group formed to address BWR/6 issues generically, submitted a position paper for the resolution of Generic Issue 7-CPB, "Rod Withdrawal Transient Analysis" (see SER Table 1.1), which changed the LRG-II original position provided in an earlier position paper. The original position stated that "a Technical Specification will be written to prohibit rod movement at indicated power levels below the low power setpoint of the Rod Control and Information System, if bypass valves are opened. The Technical Specification is intended to prevent inadvertent rod motion greater than that allowed by the Rod Withdrawal Limiter." In the June 29, 1984, position paper, the LRG-II proposed that the Technical Specification be written "to prohibit rod withdrawal at thermal power levels above the low power setpoint...." (the underlined words constitute the change proposed from the original position).

By letter dated August 13, 1984 (B. J. Youngblood to D. L. Holtzscher), the LRG-II was advised of the staff's acceptance of the wording change to be incorporated in the Technical Specification for each LRG-II plant, including Perry. However, to be applicable, each member was required to individually and formally endorse the wording change accepted by the staff. In a letter dated October 22, 1984, the applicant formally endorsed the Technical Specification wording change for Perry, indicating that the Perry draft Technical Specifications (submitted by CEI letter dated July 31, 1984) incorporate the wording changes, as appropriate. Consequently, LRG-II Generic Issue 7-CPB is considered to be fully resolved for Perry.

17 QUALITY ASSURANCE

In the process of its review of Chapter 17 of the FSAR, Amendments 13 and 14, the staff identified quality assurance (QA) organizational changes and changes in the applicant's commitments to QA regulatory guides that needed to be clarified. By letter dated September 25, 1984, the staff's questions were formally transmitted to the applicant. By letter dated October 22, 1984, the applicant provided the clarifications requested by the staff, which the applicant committed to document in a future FSAR amendment.

By letter dated December 31, 1984, the applicant submitted Amendment 15 to the FSAR, which describes a revised organization for the operations phase of the Perry plant so that Figure 17.1 of Perry SSER No. 4 is outdated in some respects. However, the reporting relationship, the organization, the responsibilities, and the authority of CEI's Nuclear Quality Assurance Department are those shown in SSER No. 4, except the Operations Quality Section is now called the Operational Quality Section. CEI's overall organization is addressed in Section 13.1 of the SER.

The staff has reviewed the applicant's above-related responses and finds them to be acceptable in satisfying its concerns. However, until the clarifications are documented in an FSAR amendment by the applicant (including the organizational changes in FSAR Amendment 15), this item will be considered a confirmatory issue. Accordingly, Confirmatory Issue (59) is being added to the list of issues in Section 1.10 of the SER by this supplement.

18 CONTROL ROOM DESIGN REVIEW

The applicant provided information about the conduct of the Perry Detailed Control Room Design Review (DCRDR) in submittals dated June 7, 1982, April 15, 1983, May 4, 1983, and June 21, 1983. No submittals were specifically identified as the DCRDR Program Plan. As a result, the staff was required to review all of the applicant's submittals to date for information about the Perry DCRDR. The staff's comments were organized around the DCRDR requirements in Supplement No. 1 to NUREG-0737 and were provided to the applicant by NRC Generic Letter 82-33 (December 17, 1982). SSER No. 4 followed the development of the Program Plan comments and summarized the staff's evaluation of the Perry DCRDR through December 1983.

The staff's Program Plan comments and report documented in SSER No. 4 noted that available information did not describe how most elements of the DCRDR would be accomplished. As a result, the applicant and the staff met to discuss those comments on February 7, 1984. The applicant subsequently responded to the staff's comments by letter dated February 29, 1984, which documented the applicant's commitment to satisfy the DCRDR requirements of Supplement No. 1 to NUREG-0737 and provided additional information about several DCRDR activities.

The February 7, 1984, meeting and the applicant's February 29, 1984, letter added to the staff's understanding of the Perry DCRDR; however, evaluation of the organization, process, and results of the DCRDR is not yet complete. As an aid in completing its evaluation, the staff is planning an early April 1985 date for a site in-progress audit of the Perry DCRDR. This audit will concentrate on DCRDR results but will also address the organization and process of the DCRDR. In addition, a further meeting with the applicant is planned preparatory to the site in-progress audit. Therefore, SER Outstanding Issue (7) continues to remain unresolved.

APPENDIX A

CONTINUATION OF CHRONOLOGY
PERRY NUCLEAR POWER PLANT, UNITS 1 and 2

January 31, 1984	Letter from applicant providing further information pertaining to SER Outstanding Issue (12), changed to Confirmatory Issue (55) in SSER No. 4.
January 31, 1984	Letter from applicant transmitting FSAR Amendment 13.
January 31, 1984	Letter from applicant partially addressing questions on containment design and isolation test provisions (Q480.50(b) and (c) and Q480.51).
February 3, 1984	NRC letter addressing deletion of home phone numbers and unlisted utility numbers from emergency plans (NRC Generic Letter 81-27 dated July 9, 1981).
February 8, 1984	NRC letter submitting two (nonprinted) advance copies of SSER No. 4.
February 14, 1984	NRC letter submitting staff report on plant environmental equipment qualification audit conducted at plant site on January 17-19, 1984.
February 16, 1984	NRC summary report of February 7, 1984, meeting with applicant to discuss Perry Detailed Control Room Design Review (DCRDR) status.
February 17, 1984	NRC letter issuing printed copies of SSER No. 4.
February 17, 1984	NRC summary report of February 10, 1984, meeting with applicant to discuss and clarify staff comments (NRC letter dated January 17, 1984) on the Perry Emergency (Onsite) Plan - SER Outstanding Issue (19).
February 23, 1984	NRC letter advising applicant that the diesel generator test program, accepted in SSER No. 4, may need to be expanded on the basis of the staff's evaluation of the Transamerica DeLaval, Inc. (TDI) diesel generator reliability - SER Outstanding Issue (24), added in SSER No. 4.
February 29, 1984	NRC letter providing Federal Emergency Management Agency (FEMA) comments on the Perry public information brochure on emergency plans.
February 29, 1984	Letter from applicant responding to staff comments on the Perry DCRDR (NRC letter dated December 23, 1983).

February 29, 1984	Letter from applicant responding to NRC Generic Letter 84-01 (January 5, 1984) concerning use of terms "important to safety" and "safety related."
March 7, 1984	Letter from applicant partially responding to staff's report on Perry environmental equipment qualification audit findings (NRC letter dated February 14, 1984).
March 9, 1984	Letter from applicant reflecting nuclear power plant experience for operating shift positions at Perry.
March 14, 1984	NRC letter requesting additional copies of full-size piping instrument drawing initially submitted by applicant in letter dated June 15, 1983 - SER Outstanding Issue (5).
March 19, 1984	NRC letter requesting information pertaining to the use of "stiff" piping clamps at Perry (Q210.15).
March 23, 1984	NRC summary report of March 12, 1984, meeting with applicant to discuss independent design verification program for Perry.
March 30, 1984	Letter from applicant requesting removal of Argonne National Laboratory from direct distribution of FSAR and FSAR amendments.
March 30, 1984	NRC letter requesting additional information relative to Mark III containment design ultimate pressure capability for Perry (Q220.30, Q220.31, and Q480.52).
April 3, 1984	NRC letter pertaining to schedules for operator and senior operator licensing examinations.
April 5, 1984	Letter from applicant submitting additional information in response to staff onsite environmental qualification audit (response to NRC letter dated February 14, 1984).
April 5, 1984	Letter from applicant scheduling seismic/dynamic qualification of electrical/mechanical equipment and concurrent pump and valve operability onsite audits by the staff.
April 6, 1984	Letter from applicant responding to Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events."
April 9, 1984	Letter from applicant providing information pertaining to pressure isolation valve leaking testing concerning SER Outstanding Issue (5) - response to NRC letter dated January 25, 1984 (Q210.14).

April 13, 1984	Letter from applicant providing additional copies of inservice testing program for pumps and valves requested by NRC letter dated March 14, 1984, pertaining to SER Outstanding Issue (5).
April 15, 1984	NRC letter resolving SER Confirmatory Issue (55), manual initiation/termination of ESF systems, introduced in SSER No. 4.
April 20, 1984	NRC letter transmitting FEMA interim report on offsite preparedness for Perry.
April 28, 1984	Letter from applicant submitting Revision 3 to the Perry Onsite Emergency Plan - SER Outstanding Issue (19).
May 7, 1984	Letter from applicant providing final response to NRC environmental qualification audit findings - a portion of SER Outstanding Issue (4).
May 7, 1984	Letter rom applicant addressing open item with respect to GE Environmental Qualification Program (NEDE-24326-P) adopted for Perry - SER Appendix H added by SSER No. 4.
May 14, 1984	NRC letter approving the Perry Security Plan, Safeguards Contingency Plan, and Security Force Training and Qualification Plan.
May 15, 1984	Letter from applicant submitting steam erosion hazards analysis report for Perry.
May 25, 1984	Letter from applicant implementing LRG-II Generic Issue 4-MEB on low-pressure coolant injection modifications in the Perry design.
May 29, 1984	Letter from applicant submitting additional information for NRC Seismic Qualification Review Team (SQRT) and Pump and Valve Operability Review Team (PVORT) audits - a portion of SER Outstanding Issue (4).
May 29, 1984	Letter from applicant providing final report on independently performed piping design review program requested by NRC.
May 30, 1984	Letter from applicant submitting Amendment 3 to the SHM License Application (10 CFR 70.34).
May 30, 1984	NRC letter requesting additional information concerning containment drywell wall structure and bypass leakage integrity for Perry (Q220.32 through Q220.36 and Q480.54).
May 31, 1984	NRC letter submitting updated status of unresolved safety issues identified in Appendix C of the SER.

June 8, 1984	Letter from applicant responding to NRC Generic Letter 84-11 (April 19, 1984) concerning inspections of BWR stainless steel piping.
June 29, 1984	Letter from applicant submitting program for environmental qualification of mechanical equipment - related to SER Outstanding Issue (4).
June 29, 1984	Letter from applicant submitting revision in the method of preoperational testing of the control room heating, ventilation, and air conditioning and emergency recirculation systems at Perry.
July 11, 1984	Letter from applicant providing clarification of LOCA-related pool dynamic loads relative to SER Outstanding Issue (9).
July 17, 1984	Letter from applicant providing additional information regarding the NRC staff Caseload Forecast conducted in March 1984.
July 18, 1984	Letter from applicant providing proposed additions to the Perry Security Plan during construction of Unit 2.
July 19, 1984	Letter from applicant submitting the Perry-specific hydrogen control program plan relative to SER License Condition (5).
July 20, 1984	Letter from applicant submitting annual financial report for 1983.
July 25, 1984	NRC letter scheduling the equipment qualification PVORT and SQRT audits at Perry site for week of August 13-17, 1984.
July 25, 1984	NRC letter submitting copies of summary forms to be used by the applicant for the SQRT/PVORT audits.
July 31, 1984	Letter from applicant transmitting draft Technical Specifications for Perry Nuclear Power Plant.
August 2, 1984	Letter from applicant submitting supplemental information for the staff PVORT/SQRT audit.
August 7, 1984	NRC letter advising of equipment selected for the equipment qualification PVORT/SQRT audit.
August 20, 1984	NRC summary report of June 19, 1984, meeting with applicant regarding Perry plant-specific responses to the hydrogen control issue - SER License Condition (5).

August 20, 1984	Letter from applicant submitting Revision 3 to the Perry Emergency (Onsite) Plan - SER Outstanding Issue (19) - responds to open items documented in SSER No. 4.
August 22, 1984	NRC letter providing staff contractor findings on the use of meteorology in emergency response at Perry.
August 22, 1984	Letter from applicant submitting FSAR Amendment 14.
August 24, 1984	Letter from applicant addressing fuel issues pertaining to SER Confirmatory Issues (11) and (12).
August 28, 1984	NRC letter requesting additional information relative to the Perry steam erosion hazards analysis.
August 28, 1984	NRC summary report of equipment qualification onsite PVORT/SQRT audit findings.
August 28, 1984	NRC Caseload Forecast Team report on projected Perry Unit 1 fuel load date.
August 28, 1984	Letter from applicant submitting Perry preservice inspection program in response to SER Confirmatory Issue (15).
August 30, 1984	NRC letter requesting additional information regarding hydrogen control for Perry relative to SER License Condition (5).
September 17, 1984	Letter from applicant addressing equipment qualification PVORT/SQRT audit findings addressed in NRC summary report dated August 28, 1984.
September 18, 1984	Letter from applicant providing clarification of staff information request relative to the Perry steam erosion hazards analysis.
September 19, 1984	Letter from applicant providing clarification of temporary addition to Perry Security Plan.
September 19, 1984	Letter from applicant responding to staff questions (NRC letter dated May 30, 1984) relative to containment drywell wall structural and bypass leakage integrity.
September 21, 1984	Letter from applicant addressing Confirmatory Issue (54), permanent dewatering system testing.
September 25, 1984	NRC letter providing results of staff review of Chapter 17 of FSAR, Amendments 13 and 14.
October 1, 1984	Letter from applicant responding to "stiff" pipe clamp questions in NRC letter dated March 19, 1984 (Q210.15).

October 2, 1984	Letter from applicant providing information clarifying Perry environmental equipment qualification program - SER Outstanding Issue (4).
October 2, 1984	Letter from applicant responding to Q420.03 through Q420.07 - effects of high energy line breaks on control systems pertaining to SER Outstanding Issue (14).
October 2, 1984	Letter from applicant providing clarification requested pertaining to safety-related systems' preoperational test requirements.
October 9, 1984	Letter from applicant endorsing LRG Instrumentation Setpoint Methodology Group plan and schedule for Perry.
October 18, 1984	NRC letter advising of errors in the numbering sequence of containment issue-related questions (series Q480) submitted with NRC letters dated March 30, May 30, and August 30, 1984.
October 22, 1984	NRC letter requesting additional information pertaining to plant conformance with TMI Action Plan Item II.D.1, testing of safety/relief valves (Q271.01 through Q271.04) - SER Confirmatory Issue (7).
October 22, 1984	Letter from applicant responding to staff questions on commitments to quality assurance regulatory quides documented in FSAR Amendments 13 and 14 (NRC letter dated September 25, 1984).
October 22, 1984	Letter from applicant endorsing staff-approved Technical Specification wording changes as applicable to Perry - LRG-II Generic Issue 7-CPB, "Rod Withdrawal Transient Analysis."
October 25, 1984	Letter from applicant responding to staff contractor questions/findings (NRC letter dated August 22, 1984) concerning use of meteorology in emergency response at Perry.
October 29, 1984	Letter from applicant clarifying Revision 3 of emergency plans (OM-15A) - open items documented in Section 13.3 of SSER No. 4.
October 31, 1984	NRC letter providing additional staff questions concerning Chapter 13 of FSAR Amendment 14.
October 31, 1984	Letter from applicant responding to TDI diesel generator questions submitted by NRC letter dated December 23, 1983.

November 1, 1984	Letter from applicant providing program for resolution of effects of local encroachments in the suppression pool on swell impact loads - related to SER Outstanding Issue (8).
November 13, 1984	NRC letter accepting revised test method for preoperational testing of control room ventilation system.
November 14, 1984	NRC letter requesting information on containment purge and vent valve operability provisions of TMI Action Plan Item II.E.4.2(f) - SER Confirmatory Issue (23).
November 20, 1984	Letter from applicant responding to NRC letter dated October 22, 1984, forwarding questions on the applicability of generic safety/relief valve test results to Perry as reported in GE Technical Report NEDE-24988-P - SER Confirmatory Issue (7).
November 20, 1984	NRC letter indicating acceptability of Perry compliance with NUREG-0612 relative to the control of heavy loads (Phase I) - SER License Condition (18).
December 6, 1984	NRC report of November 15, 1984, meeting with applicant to discuss Perry containment bypass leakage design and LOCA-related pool dynamic loads (SER Outstanding Issue (9)).
December 10, 1984	NRC letter advising applicant of TDI diesel generator reliability verification required for licensing Perry.
December 11, 1984	NRC letter submitting interim report on Perry comformance with the emergency capability guidelines of Regulatory Guide 1.97, Revision 2.
December 14, 1984	Letter from applicant submitting revised preservice inspection program for Perry - SER Confirmatory Issue (15).
December 19, 1984	NRC letter transmitting Idaho National Laboratory trip report concerning site audit of seismic and dynamic qualification of safety-related equipment.
December 19, 1984	Letter from applicant providing additional information related to Perry compliance with TMI Action Plan Item II.E.4.2(f) on purge and vent valve operability - SER Confirmatory Issue (23).
December 31, 1984	Letter from applicant submitting FSAR Amendment 15.
January 4, 1985	Letter from applicant transmitting Revision 1 to draft Technical Specifications (letter dated July 31, 19C4).

January 4, 1985	Letter from applicant providing additional information concerning Perry meteorological information dose assessment system requested by NRC letter dated August 22, 1984.
January 10, 1985	Letter from applicant providing procedures generation package utilizing guidelines from NRC letter dated May 6, 1983 - SER Outstanding Issue (21).
January 10, 1985	Letter from applicant submitting summary report of the Detailed Control Room Design Review - SER Outstanding Issue (7).
January 14, 1985	NRC letter addressing resolution of Construction Appraisal Team concern about containment drywell wall structural and leaktightness integrity.
January 14, 1985	Letter from applicant responding to staff's questions (in NRC letter dated October 31, 1984) concerning Chapter 13 changes reflected in FSAR Amendment 14.
January 14, 1985	Letter from applicant updating information previously submitted (by letters dated January 14, 1983, and November 1, 1983) regarding GE inadequate core cooling report implemented by the BWROG - SER License Condition (4).
January 16, 1985	NRC letter requesting an update of the operator staffing experience tables initially submitted in March 1984.
January 16, 1985	Letter from applicant further clarifying Emergency Plan (Rev. 3) information in letters dated April 28, 1984, August 20, 1984, and October 29, 1984 - SER Outstanding Issue (19).
January 17, 1985	Letter from applicant providing TDI diesel generator reliability issue program plan requested by NRC letter dated December 10, 1984 - SER Outstanding Issue (24).
January 22, 1985	NRC letter requesting additional information regarding potential thermal gradient stresses in safety/relief valve discharge piping (followup to Q210.10 submittal by NRC letter dated January 21, 1983).
January 24, 1985	NRC letter requesting additional information concerning the plant safety parameter display system.
January 24, 1985	NRC letter submitting Detailed Control Room Design Review Summary Report establishing site in-progress audit for week of April 8-12, 1985 - SER Outstanding Issue (7).

APPENDIX B

REFERENCES*

Advisory Committee on Reactor Safeguards, Report No. 0938, August 11, 1981.

Cleveland Electric Illuminating Company, "Final Safety Analysis Report for the Perry Nuclear Power Plant, Units 1 and 2" (Docket Nos. 50-440 and 50-441), through Amendment 14, August 1984.

Code of Federal Regulations, Title 10, "Energy" (10 CFR), and Title 44, "Emergency Management and Assistance" (44 CFR) (includes General Design Criteria (GDC)).

General Electric Company, Report SLI-8211, "Review of Reactor Water Level Measurement Systems," July 1982.

---, Report SLI-8218, "Inadequate Core Cooling Detection in BWRs," December 1982.

---, Topical Report NEDE-21175-P, "BWR Fuel Assembly Evaluation of Combined SSE and LOCA Loadings," Amendment 3, July 1982. Proprietary information. Not publicly available.

---Topical Report NEDE-24326-P, "General Electric Environmental Qualification Program," Rev. 1, January 1983. Proprietary information. Not publicly available.

Keller, H., "Erosion-Corrosion in Wet Steam Turbines," Kraftwork AG, Mulheim VGB Kraftworkstachnik, Vol. 54, No. 5, May 1984.

Letter, May 7, 1982, C. A. Cameron (GE) to W. R. Mills (NRC), "Kuosheng Incore Instruments Tube Break."

---, May 18, 1982, R. Artigas (GE) to R. Tedesco (NRC), "LPCI Flow Deflector Redesign for BWR/6."

---, September 3, 1983, D. L. Holtzscher (LRG-II) to T. A. Novak (NRC), "Position on 4-MEB Regarding Kuosheng Incore Instrument Tube Break."

---, September 9, 1983, D. L. Holtzscher (LRG-II) to J. J. Stefano (NRC), "Submittal of Position Papers for Licensing Review Group-II Issues" (includes Issue 4-MEB and other issues).

*All correspondence between the applicant and the NRC staff referenced in this supplement is listed in Appendix A of the SER and its supplements on a continuing chronological basis.

---, December 14, 1983, B. J. Youngblood (NRC) to D. L. Holtzscher (LRG-II), "Acceptance of LRG-II Proposed Solution for Generic Issue 4-MEB - LPCI Modifications."

---, May 15, 1984, T. M. Novak (NRC) to J. F. Carolan (ISMG), "Transmittal of NRC Staff Report on Setpoint Methodology for General Electric Supplied Protection System Instrumentation."

---, June 29, 1984, J. F. Carolan (ISMG) to T. M. Novak (NRC), "Action Plan To Answer NRC Staff Concerns on Setpoint Methodology for General Electric Supplied Protection System Instrumentation."

---, June 29, 1984, D. L. Holtzscher (LRG-II) to J. J. Stefano (NRC), "Submittal of Position Papers for Licensing Review Group-II Issues" (includes Issue 7-CPB and other issues).

---, July 23, 1984, B. J. Youngblood (NRC) to J. F. Carolan (ISMG), "Acceptance of Action Plan To Answer NRC Staff Concerns on Setpoint Methodology for General Electric Supplied Protection System Instrumentation."

---, August 13, 1984, B. J. Youngblood (NRC) to D. L. Holtzscher (LRG-II), "NRC Staff Acceptance of the Licensing Review Group (LRG)-II Position for Generic Issue 7-CPB, 'Rod Withdrawal Transient Analysis.'"

U.S. Nuclear Regulatory Commission, Generic Letter, "Control of Heavy Loads - NUREG-0612," December 22, 1980.

---, Generic Letter 81-03, "Implementation of NUREG-0313, Rev. 1, 'Technical Report on Material Selection and Processing Guidelines for BWR Coolant Presure Boundary Piping,'" February 26, 1981.

---, Generic Letter 82-33, "NUREG-0737, Supplement 1," December 17, 1982.

---, Generic Letter 83-24, "TMI Task Action Item I.G.1, Special Low Power Testing and Training Recommendations for BWRs," June 29, 1983.

---, Generic Letter 84-23, "Reactor Vessel Water Level Instrumentation in BWRs," October 26, 1984.

---, NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Rev. 1, July 1980.

---, NUREG-0484, "Methodology for Combining Dynamic Responses," Rev. 1, May 1980.

---, NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants - Resolution of Generic Technical Activity A-36," July 1980.

---, NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Rev. 1, November 1980.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," Supp. 1, December 1982.

---, NUREG-0800 (formerly NUREG-75/087), "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants," Rev. 2, July 1981 (includes Branch Technical Positions).

---, NUREG-0887, "Safety Evaluation Report Related to the Operation of Perry Nuclear Power Plant, Units 1 and 2," May 1982; Supp. 1, August 1982; Supp. 2, January 1983; Supp. 3, April 1983; Supp. 4, January 1984.

---, NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants (Technical Findings Relevant to Unresolved Safety Issue A-1)," Rev. 1, March 1984.

---, Office of Analysis and Evaluation of Operating Data (AEOD), AEOD/E4-16, "Erosion in Nuclear Power Plants," by E. J. Brown, June 11, 1984.

---, Office of Inspection and Enforcement (IE) Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment," January 14, 1980, and supplements dated February 29, September 30, and October 24, 1980.

---, IE Bulletin 79-02, "Pipe Support Base Plate Design Using Concrete Expansion Anchor Bolts," March 8, 1979.

---, IE Bulletin 79-22, "Possible Leakage of Tubes of Tritium Gas Used in Timepieces for Luminosity," September 5, 1979.

---, IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," November 30, 1979.

---, IE Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Controls," March 3, 1980.

---, IE Information Notice 79-22, "Qualification of Control Systems," September 19, 1979.

---, IE Information Notice 82-22, "Failures in Turbine Exhaust Lines," July 9, 1982.

---, IE Information Notice 82-23, "Main Steam Isolation Valve (MSIV) Leakage," July 16, 1982.

---, IE Information Notice 83-28, "Criteria for Protective Action Recommendations for General Emergencies," May 4, 1983.

APPENDIX C

UNRESOLVED SAFETY ISSUES

C.5 Discussions of New USIs as They Relate to Perry Units 1 and 2

Task A-1 Waterhammer

In Section C.5, Appendix C, of the SER, the staff identified, as an unresolved safety issue (USI), waterhammer events, which introduce large hydraulic loads, or pressure pulses, into a fluid system and are the result of rapid condensation of steam pockets. It was stated that the resolution of safety concerns related to the waterhammer issue was being pursued by the staff under Task A-1.

NUREG-0927, Revision 1, published in March 1984, is a report of the staff's technical findings relevant to USI Task A-1 that were derived from studies of reported waterhammer occurrences and underlying causes, and provides key insights into means to minimize or eliminate further waterhammer occurrences. The major conclusions reached are that the frequency and severity of waterhammer occurrences can be and, to some extent, have been significantly reduced through design features incorporated in plant designs, including that of Perry, and that the current potential for significant damage as a result of waterhammer events is less than it was in earlier plant designs. As such, with the publication of NUREG-0927, Revision 1, this USI is no longer at issue for Perry and is considered to be resolved.

Task A-46 Seismic Qualification of Equipment in Operating Plants

In Section C.5, Appendix C, of the SER, the staff reported that Perry was designed on the basis of current seismic design criteria, and commitments for seismic equipment qualification were in accordance with the latest ASME Codes and standards. Since the scope of Task A-46 is limited to dealing with seismic qualification of equipment in operating plants, the issue related to Task A-46 is not applicable to Perry and is hereby deleted.

C.6 Reference

U.S. Nuclear Regulatory Commission, NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants. Technical Findings Relevant to USI A-1," Rev. 1, Mar. 1984.

APPENDIX E

NRC STAFF CONTRIBUTORS AND CONSULTANTS

NRC STAFF

Name	Title	Branch
G. Bagchi	Section Leader	Equipment Qualification
H. Brammer	Section Leader	Mechanical Engineering
W. Brooks	Reactor Physicist	Core Performance
J. Halapatz	Materials Engineer	Materials Engineering
P. Hearn	Mechanical Engineer (auxiliary systems)	Auxiliary Systems
S. Hou	Sr. Mechanical Engineer	Mechanical Engineering
J. Joyce	Sr. Reactor Engineer (instrumentation)	Instrumentation and Control Systems
K. Leu	Sr. Structural Engineer	Structural and Geotechnical
R. Li	Mechanical Engineer	Mechanical Engineering
R. Manili	Plant Protection Analyst	Power Reactor Safeguards*
J. Mauck	Reactor Engineer (instrumentation)	Instrumentation and Control Systems
J. Norton	Inspector	Region III
D. Perrotti	Emergency Preparedness Engineer	Emergency Preparedness**
R. Romney	Equipment Qualification Engineer	Equipment Qualification
D. Serig	Human Factors Engineer	Human Factors Engineering
A. Serkiz	Task Manager (USI)	Generic Issues
A. Singh	Mechanical Engineer (auxiliary systems)	Auxiliary Systems
J. Spraul	Sr. Quality Assurance Engineer	Quality Assurance**
G. Stahley	Hydraulic Engineer	Environmental and Hydrologic Engineering
N. Su	Task Manager (USI)	Generic Issues
S. Sun	Nuclear Engineer	Core Performance
H. Walker	Mechanical Engineer	Equipment Qualification
R. Wright	Sr. Equipment Qualification Engineer	Equipment Qualification
L. Yang	Structural Engineer	Structural and Geotechnical

CONSULTANTS

Name	Organization
T. Bridges	EG&G, Idaho
C. Kido	EG&G, Idaho
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J. Sekot	EG&G, Idaho
J. Singh	EG&G, Idaho
T. Stickley	EG&G, Idaho
A. Udy	EG&G, Idaho

*Office of Nuclear Materials Safety and Safeguards.
**Office of Inspection and Enforcement.

APPENDIX G

ERRATA TO THE SAFETY EVALUATION REPORT AND SUPPLEMENTAL SAFETY EVALUATION REPORTS

Page	Section/Table	Change
A-6 SER	Appendix A	Add "in response to Confirmatory Issue (22)" in description of April 26, 1982, letter between applicant and NRC staff.
6-1 SSER No. 2	6.2.1.9	In first line of third paragraph, change "June 7, 1982" to "April 26, 1982."
6-4 SSER No. 2	6.3.1.3	Under the subparagraph "Requirements," delete the third sentence in its entirety which begins, "Air or nitrogen...."
6-5 SSER No. 2	6.3.1.3	Under the subparagraph "Design Description," line 9, delete "four" and substitute "five" before the word "actuations."
A-2 SSER No. 2	Appendix A	Delete "June 7, 1982, Letter from applicant responding to SER Confirmatory Issue (22)."

APPENDIX I

PERRY SQRT VISIT REPORT

EGG-EA-6733

PERRY
SQRT VISIT REPORT

T. L. Bridges
M. J. Russell
J. N. Singh

Published November 1984

EG&G Idaho, Inc.
Idaho Falls, Idaho 83415

Prepared for the
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Contract No. DE-AC07-76ID01570
FIN No. A6415

ABSTRACT

EG&G Idaho is assisting the Nuclear Regulatory Commission in evaluating Cleveland Electric Illuminating Company's program for the dynamic qualification of safety related electrical and mechanical equipment for the Perry Nuclear Power Generating Station. Applicants are required to use test or analysis or a combination of both to qualify equipment, such that its safety function will be ensured during and after the dynamic event, and provide documentation. The review, when completed, will indicate whether an appropriate qualification program has been defined and implemented for seismic Category I mechanical and electrical equipment which will provide reasonable assurance that such equipment will function properly during and after the excitation due to vibratory forces of the dynamic event.

CONTENTS

TABLES

SUMMARY

A seismic qualification review team (SQRT) consisting of engineers from the Equipment Qualification Branch of the Nuclear Regulatory Commission and Idaho National Engineering Laboratory made a site visit to the Perry Nuclear Power Plant of Cleveland Electric Illuminating Company located at Perry, Ohio. They observed the field installation and reviewed the qualification reports for twenty-two selected pieces of seismic category I electrical and mechanical equipment and their supporting structures. Some equipment specific and certain general concerns were identified for which additional information is needed in order for the SQRT to complete the review. These are referred to as open items. The review indicated that the equipment was adequately qualified for the dynamic environment pending resolution of the open items.

1. INTRODUCTION

The Equipment Qualification Branch (EQB) of the Nuclear Regulatory Commission (NRC) has the lead responsibility in reviewing and evaluating the dynamic qualification of safety related mechanical and electrical equipment. This equipment may be subjected to vibration from earthquakes and/or hydrodynamic forces. Applicants are required to use test or analysis or a combination of both to qualify equipment essential to plant safety, such that its function will be ensured during and after the dynamic event. These pieces of equipment and how they meet the required criteria are described by applicants in a Final Safety Analysis Report (FSAR). On completion of the FSAR review, evaluation and approval, the applicant receives an Operating License (OL) for commercial plant operation.

A Seismic Qualification Review Team (SQRT) consisting of engineers from the EQB of NRC and Idaho National Engineering Laboratory (INEL), made a site visit to the Perry Nuclear Power Plant of Cleveland Electric Illuminating Company at Perry, Ohio, from August 14 through August 17, 1984. The purpose of the visit was to observe the field installation, review the equipment qualification methods, procedures (including modeling technique and adequacy), and documented results for a list of selected seismic Category I mechanical and electrical equipment and their supporting structures. This report, containing the review findings, indicates which of the items are qualified and require no additional documentation. It also identifies some equipment and certain general concerns for which additional information is needed in order for the SQRT to complete the review. These are referred to as open items. The applicant is to further investigate and provide additional documentation to resolve these issues.

Table 1 contains a list of personnel who attended the site visit. Subsequent sections of this report give a brief overview and identify the concerns, followed by the findings, for the selected seismic Category I equipment.

2.1 ELECTRICAL PROTECTION ASSEMBLIES (NSSS-1)

The electrical protection assemblies are wall mounted electrical boxes and associated devices. They measure 16 in. x 20 in. x 8 in. and weigh approximately 150 lb each. These assemblies were manufactured by General Electric Co. with model No. GE-DWG-914E175. All eight assemblies are located at the 620 ft elevation of the control complex. Each is attached to the wall with four 3/8 in. diameter bolts. Two inch diameter rigid conduits attach to the boxes at both the bottom and top. Each assembly must perform its 1E function during both hot standby and cold shutdown conditions.

The assemblies were purchased to GE purchase specification 21A3120. These items were qualified for dynamic loading by dynamic testing performed by GE, documented by test Report DRF C71-00044. The testing consisted of low level resonance search and biaxial, random, multifrequency dynamic testing. The assembly lowest natural frequencies were determined to be 54.5 Hz side/side (s/s), 61 Hz front/back (f/b), and 54.5 Hz vertical (v). The required test spectra zero period accelerations (ZPA) for the operating basis earthquake (OBE) are 0.213 g s/s and f/b, and 0.36 g v. The corresponding values for safe shutdown earthquake (SSE) are 0.33 g and 0.58 g. The test input ZPA values were as follows: OBE: s/s 5.4 g, f/b 8.0 g, v 7.0 g; SSE: s/s 11.0 g, f/b 10.07 g, v 8.9 g. The specimen was subjected to 10 OBE and 4 SSE tests. Half of each series was with the test motion in-phase and half out-of-phase. The specimen was test mounted flat rather than vertically. Acceptability of this mounting is justified because the test input levels are significantly higher than the required dynamic loading in combination with gravity loading. Operability of the assembly was verified during and after the dynamic testing.

Based on our observation of the field installation, review of the qualification reports and the responses by the applicant to our questions, the electrical protection assemblies are adequately qualified for the defined seismic and hydrodynamic loads.

2.2 ROD POSITION MULTIPLEXER CABINET (NSSS-2)

The rod position multiplexer cabinet (model No. H22-P071) was supplied by General Electric Company (GE). It measures 72 x 53 x 30 in. and weighs approximately 1500 lb. This cabinet is located at the 620 ft elevation of the containment building. The mounting consists of twelve 9/16 in. bolts to the floor.

This equipment is qualified by tests performed on a similar unit by General Electric Company. Seismic loads are considered in the qualification. The pertinent design specification for qualification requirements are in: GE Drwg 22A3746--Design Spec. Local Instrument Panel EDLPL865E133BAG001, Rev 6; Assy 865E133BA, Rev 2. The qualification details are in BWR 6 Rod Control and Information System Panels; GE DRF No. A00-794-8, dated March 18, 1980.

The rod position multiplexer cabinet is required to transmit correct rod position information or to fail safely during normal plant operation and after a seismic event. In any event, the multiplexer cabinet must not fail in a way that violates divisional separation of electrical systems.

Reportedly, the equipment was tested up to high frequency, but the data was reduced only up to 23 Hz. Due to the loss of high frequency data, GE did a preliminary failure mode analysis. GE stated that this analysis indicates that none of the failure modes identified can prevent the position multiplexer from failing safely or can have a detrimental impact on the bus supplying power to this equipment. Thus, this unit is not a safety related item. The utility is not convinced of the argument and wants GE to justify their contention. GE is in the process of a detailed analysis of the failure modes. In case it turns out that this equipment is safety related, the item would be requalified according to Perry 1 requirements, including high frequency testing.

Based on the responses of the applicant to our questions, it is concluded that there is a mechanism in place (work is in progress) which would assure the seismic qualification of the rod position multiplexer cabinet to a satisfactory level. The utility has agreed to inform the NRC when qualification is completed.

2.3 CONTROL ROD DRIVE VENT VALVE (NSSS-3)

The control rod drive vent valve is a one inch air operated globe valve. It was supplied by ITT Hammer Dahl with model No. 522FRR62HAZ9. The valve is line mounted in a one inch schedule 160 pipe line at elevation 642 ft inside the containment. The piping has vertical supports within 8 in. of either side of the valve. Horizontal support of the valve is provided near the vertical center of gravity of the valve-operator combination. This valve is normally open and is required to close during scram conditions. It is required for both hot standby and cold shutdown conditions. The valve closes upon the loss of operator air.

The valve was purchased to design specifications 22A6924, 22A6924AA, and 23A1331. It was qualified for seismic and hydrodynamic loading by testing performed by Wyle Laboratories, documented by test Report No. 58840 Rev. B, dated February 5, 1984. The test valve nozzles were welded to book-end shaped plates which were rigidly attached to the test table. The valve-operator combination natural frequencies were determined to be 20 Hz s/s, 60 Hz f/b, and rigid vertically. The specimen was tested using random, independent biaxial test motion. The testing consisted of 5 OBE and 1 SSE tests of 30 seconds duration each. Since the test mounting was more rigid than normal valve line mounting, the input test motion ZPA was set above the floor required response spectra peak. The valve and operator operated without any anomalies during and after the testing. Valve stroke times were within the allowable 30 seconds.

Based on our observation of the field installation, review of the qualification reports and the responses by the applicant to our questions, the control rod drive vent valve is adequately qualified for the defined seismic loads.

2.4 SQUIBB VALVE (NSSS-4)

The squibb valve (MPL No. 1C41F0004B, model No. 1832-159-01) was supplied to design specification Nos. 21A9370AB rev. 7 and 21A9370 rev. 6 by Conax. It is located in the reactor building at the 644 ft elevation. It was installed between flanges of the piping system with four 1 in. bolts. The valve weighs 40 lb and is 4.5 in. long by 7 in. diameter. The valve is part of the standby liquid control system. Its function is to provide leak tight isolation of the system until it is fired, at which time it allows injection of poison into the primary coolant.

The valve was qualified by test (General Electric report No. NEDC-30207, Environmental Qualification Report, SLCS Explosive Valve, October 1983). The fundamental natural frequency of greater than 100 Hz was calculated for the valve. It was subjected to a full test series. Vibration, thermal and radiation aging was applied before the seismic test. This included fifteen minutes of sine beat testing in each axis at an input acceleration value of 4.5 g, which would address aging due to hydrodynamic loads. Mounting for vibration testing was identical to the field mounting. Seismic qualification consisted of sine beat testing in three orthogonal axes. For each axis, five OBE tests were performed at an acceleration of 4.5 g and in the frequency ranges 5-20, 45-60 and 90-100 Hz. One SSE test was performed in each axis at an acceleration value of 6.75 g. These input acceleration values envelope the RRS ZPA values of 0.3 g horizontal and 0.2 g vertical for OBE, and 0.45 g horizontal and 2.0 g vertical for SSE.

Based on our observation of the field installation and review of the qualification documents, the squibb valve is adequately qualified for the prescribed loads.

2.5 REACTOR CORE ISOLATION COOLING STEAM TURBINE ASSEMBLY (NSSS-5)

The reactor core isolation cooling (RCIC) steam turbine assembly (model No. GS-2) was supplied by Terry Corporation. It is located in the auxiliary building at the 599 ft elevation and has a rated horsepower of 825 at 4550 revolutions per minute. The maximum inlet and outlet pressures are prescribed as 1250 psig at 575°F and 165 psig. The maximum governor valve travel is specified as 7/8 in. This base mounted, single wheel steam turbine measures about 7 ft long by 6 ft wide by 5 ft high and weighs approximately 5000 lb. The mounting consists of six 1 in. bolts. It drives the RCIC pump, providing cooling water to the reactor vessel over a pressure range of 1150 psig to 150 psig. The purchase specification from GE to Terry Turbine is in document No. 21A9526 rev. 2 dated July 5, 1973, (generic) and the Perry specification is: 21A9526AE of March 15, 1974.

The qualification of the turbine assembly is based on both analysis and test. The qualification documents are: Environmental Qualification Report E/L 20452, Rev. 1, dated April 21, 1980, prepared by Wyle Laboratories and Terry Corporation and reviewed by General Electric Company. Other supporting documents are:

S.N.	Document Identification	Revision or Date	Title or Subject
1	VPF 2757-74-1	7149	Turbine Stress Analysis
2	VPF 2757-34-2	7002	Stop Valve Stress Analysis
3	VPF 2757-35-1	7030	Seismic Analysis

The function of the pump driven by the turbine is to supply makeup water in three cases:

(i) loss of feed, loss of main condenser, etc.

(ii) control rod drop accident, and

(iii) anticipated transient without scram event.

Qualification Testing

The specimen was the same type of turbine assembly. The mounting consisted of the assembly base plate bolted to a rigid metal fixture which was in turn welded to the test table. This simulated the actual in-service mounting condition as closely as practical. Two RCIC turbine assemblies, Model GS-2, were subjected to dynamic qualification test programs. In the first program, the unit had a set of appendages, a particular support condition, and was not aged. This program showed some undesirable vibration characteristics and anomalies were detected. Subsequently, a second test program was commissioned which had the same model turbine assembly (GS-2), a slightly different set of appendages, modified support conditions, and was aged prior to dynamic testing. These appendages and support modifications were based on the lessons learned from the first test program. The following discussion pertains to the second test program.

The turbine assembly was subjected to the following test sequence:

Resonance search:	First horizontal axis and then vertical axis
Qualification tests:	Five OBE level and one SSE level tests performed
Rotate the specimen:	90 degrees about the vertical axis
Resonance search:	Second horizontal axis

Qualification tests:	Five OBE level and one SSE level tests performed.

The resonance search tests were performed with a 0.2 g single axis sine sweep input between 1 Hz to 60 Hz, at a sweep rate of one octave per minute. The following natural frequencies were detected:

s/s	f/b	v
17, 24 Hz	16, 22 Hz	18, 33 Hz

The qualification tests consisted of two simultaneous, but independent random signals which produced phase-incoherent horizontal and vertical motion. The amplitude of each one-third octave bandwidth was independently adjusted in each axis until the TRS enveloped the generic RRS. The resulting motion was analyzed by a response spectrum analyzer at two percent damping and plotted at one-sixth octave interval over the frequency of 1 to 100 Hz. The generic RRS envelops the Limerick site required response spectrum for the floor where the unit is located. The dynamic simulation tests were each of 30 seconds duration. Operability of the turbine assembly (startup, steady state operation, and shutdown) was demonstrated during and after the tests. Two anomalies were detected during the test. Discussion of the anomalies and their disposition follows.

1. Turbine Trip: The turbine tripped during the SSE level run.

Discussion: The turbine trip was attributed to loose flange bolting at the interface of the governor valve and the turbine casing. This resulted in highly amplified vibration at the trip and throttle valve, causing the trip latching mechanism to separate. The bolting was retorqued and the test was repeated successfully. Six additional tests were performed with no further problems.

Disposition: The dynamic portion of the test specification defines the acceptability of tightening the hardware after each test. Furthermore, the maintenance portion of the turbine instruction manual defines the necessity for verifying bolt and stud-nut torque condition after every seismic event. The turbine trip occurred on the sixth test of the series. There had been no retorquing prior to this test, and none was required during the subsequent seven tests. It is, therefore, concluded that the design is adequate, and no corrective action is necessary.

2. Lube oil piping vibration: During the resonance search and the first OBE level testing, excessive displacement was observed in the oil piping to the coupling end bearing.

Discussion: The resonance search revealed a natural frequency of 15 to 20 Hz for the oil piping to the coupling end bearing. The first OBE level test revealed excessive displacement. This is critical piping. Its integrity is required for proper functioning of the turbine. A support bracket was added to stiffen the piping assembly. Subsequent tests were conducted without incident.

Disposition: The oil piping on all turbine assemblies will be inspected for support adequacy, with additional support added where necessary.

The lessons learned from the test program (on the unaged turbine assembly) included several areas where structural improvement was required. They are:

a. stiffness of the latching lever spring on the turbine trip and throttle valve,

b. dowel pin adequacy for the turbine coupling end pedestal,

c. positive locking of the turbine pedestal bolts, and

d. support adequacy for the turbine auxiliary piping.

These improvements were implemented on the second turbine assembly (the aged assembly test) prior to the start of its qualification tests, with the exception of the turbine auxiliary piping. This is because the piping supports are installation-unique to a degree. The results of this test program demonstrated the adequacy of the structural upgrade activity. The unit in Perry 1 has been upgraded.

Qualification Analysis

The original turbine analysis conservatively demonstrated the pressure integrity of the assembly. Using these calculations and the accelerometer data recorded during the qualification test program, detailed calculations were performed to define the nozzle load capabilities. A typical set of force/moment loads were applied to the turbine inlet and exhaust nozzle during the test. In brief, the structural adequacy of the assembly was ascertained as follows:

a. stresses due to pressure were calculated,

b. stresses due to dynamic loads were calculated from the acceleration data, and

c. the rest of the strength was available for nozzle loads.

Applicability of the Qualification Program

The test specimen and the installed product are both complete RCIC turbine assemblies. However, they were manufactured at substantially different times. A detailed similarity evaluation was performed by the applicant to identify the differences between the tested and installed unit. The upgrading and design differences to be implemented for the Perry unit to be fully qualified are as follows:

1. In the first qualification test program, #8 taper pins were used for coupling-end alignment. One of these pins failed after 31 tests (accumulated test time of approximately 15 minutes).

The turbine for the second test program (which was a success) used #9 taper pins. It also had lock plates for pedestal bolting. Therefore, the Perry turbine needed upgrading with #9 taper pins and lock plates for the pedestal bolting. This has been done in the field.

2. During the first seismic qualification test program, the initial test activity resulted in inadvertent, unacceptable closure of the trip and throttle valve. The original latching spring was replaced with one having a higher spring coefficient. The operability of the solenoid trip mechanism and the mechanical overspeed trip mechanism were verified after installation of the stiffer latch spring, and proved to be acceptable. The seismic qualification test program was then successfully completed.

   As a result the field installation required the following corrective action. The latch spring was to be removed from the trip and throttle valve assembly, and its spring constant measured, which should be 25 lb/in., ±10%. If the installed spring did not satisfy this value, it was to be replaced. The spring "load" in the valve latched position was to be 32.5 lb. This check has been done in the field, according to the applicant.

3. Each RCIC turbine installation has somewhat of a unique piping arrangement. For turbine oil piping adequacy, therefore, the Perry as-installed piping has been reviewed and adequate supports provided.

Aging evaluation of the nonmetallic mechanical components of the assembly has not been performed. However, there is a program in place (presented by the applicant) which, when completed, will alleviate this problem.

Based on observation of the field installation, review of the qualification documents and responses of the applicant, the RCIC turbine assembly will be seismically qualified when the program is completed.

2.6 TERMINATION CABINET (NSSS-6)

The termination cabinet (MPL No. 1H13P0702, no model No.) was supplied to the standard Perry NSSS design specification by General Electric. It is located in the control complex at the 654.5 ft elevation. The cabinet is attached to the floor with eighteen 5/8 in. bolts. It weighs approximately 2400 lb and measures 96 W x 102 H x 36 D in. The cabinet is a control room panel, where it serves as an interface between control room circuitry and electrical cables entering the control room.

Qualification of the cabinet was based on a similarity argument between this cabinet and a series of cabinets qualified by test (General Electric report No. GE DRF A00-794-5-1, Seismic Qualification Test Report, October 1, 1980). The tested cabinets were subjected to a series of multifrequency (random) tests on a biaxial test table. Test mounting was with sixteen 5/8 in. bolts. Testing was performed in two positions, with in-phase and out-of-phase inputs in each position. Five OBE tests followed by an SSE test were performed. TRS for all testing enveloped the Perry RRS. Since the spectra did not include hydrodynamic effects, there was a concern that a cabinet qualified to this methodology could be mounted in an area of the plant subject to hydrodynamic excitation. A check was made, and all the cabinets of interest were found to be located in the control complex, which is isolated from hydrodynamic loading.

GE analyzed the test results. Variations in cabinet response as a function of cabinet size, aspect ratios, mass, and mass distribution were established. All cabinets were of similar construction. Results of the study were used to demonstrate similarity of the termination cabinet with those tested.

Based on inspection of the field installation, review of the qualification documents, and the applicant's response to questions, the termination cabinet is adequately qualified for the prescribed loads.

2.7 CONTROL ROOM PANEL (NSSS-7)

The control room panel (model No. H13-P680) was supplied by General Electric Company (GE). It measures 240 x 33 x 63.5 in. and weighs about 7200 lb. It is located in the control room at the 654 ft 6 in. elevation. The mounting consisted of welding at the corners to the floor channel. Seismic loads are considered in the qualification. The qualification details are contained in report: Test Report H13-P680 Prototype Center Enclosure Compact Principal Plant Control Console, GE DRF A00-1138.

The qualification is based on the tests performed on a dynamically similar panel (minute dimensional difference). Test mounting consisted of clamps at angle supports where welding in the field was specified. The required ZPA for the location were:

	s/s	f/b	v
OBE:	0.43 g	0.43 g	0.40 g
SSE:	0.64 g	0.64 g	0.60 g

A low level resonance search indicated the following natural frequencies:

s/s	f/b	v
24 Hz	19.5 Hz	28 Hz

The qualification test consisted of biaxial (independent) random input. The test was done with in-phase and out-of-phase inputs and then repeated after the equipment was rotated 90 degrees about the vertical axis. TRS were generated. The TRS envelope the RRS satisfactorily. Four percent damping was used for both RRS and TRS.

Panel structural integrity was maintained and the class 1E devices functioned as required during and after the test. There were five OBE and one SSE level tests performed.

There was a difference between the field and the laboratory supports. In the laboratory mounting, welding at each angle support was specified (where it was clamped), but in the field it is only welded at the corner. The applicant has committed to provide a justification for the discrepancy.

During the review of the documentation it was discovered that there were unqualified items on panels H13-P883, P884, P885 but not on H13-P680. This was pointed out to the applicant. The applicant is required to address these items.

Based on observation of the field installation, review of the qualification reports and the responses of the applicant, the control room panel is adequately qualified for seismic loads pending resolution of the mounting difference and the unqualified items.

2.8 BENCHBOARD (NSSS-8)

The benchboard (control room panel, MPL No. 1H13P0870, model No. H13-P870) was supplied to the standard Perry NSSS design specification by General Electric Co. It is located in the control room, at the 654.5 ft elevation. The benchboard was attached to the floor with eighty-four 1/2 in. bolts. It weighs approximately 2400 lb and measures 252 W x 36 D x 90 H in. The benchboard is a control room panel, where it serves as a support for electrical instruments such as displays, indicators and switches.

The benchboard was qualified by test (General Electric report No. GE DRF A00-1138, H13-P870 Seismic Test, Index I). It was mounted to the test table in a fashion identical to the field mounting. A series of multi-frequency (random) biaxial tests were performed. Testing was performed in two positions, with in-phase and out-of-phase inputs in each position. Five OBE tests were performed, followed by one SSE test. All TRS enveloped the appropriate Perry RRS. Panel structural integrity was demonstrated and all mounted class 1E devices functioned as required.

Although not required for qualification in this case, fundamental natural frequencies of 14.0 Hz (s/s) and 17.5 Hz (f/b) were established during testing. These compared to frequencies of 17.5 Hz (s/s) and 16.8 Hz (f/b) obtained from in-situ testing.

Generally, not all cabinet mounted instruments at Perry were qualified simultaneously with their cabinets, as was done with this benchboard. Therefore a check was made on a cabinet mounted relay which was qualified separately from its cabinet. The device chosen was a relay (MPL No. D17-74-E) mounted in cabinet H13-P872. The G.E. qualification document number for this device is DRF A00-1084, Index 212. The document was found to be incomplete, with test procedure QTP-GP-1 missing. The test procedure was located in the San Jose Office of GE, but a copy of it should have been included with the qualification document. This was classified as an isolated incident, since all other qualification files reviewed were complete. Test results found in the document indicated that single frequency fragility testing was performed. This included a determination of fundamental natural frequency (rigid). The failure mode of the relay was determined to be failure of the attachment to the table, and not a malfunction of the relay. This occurred at 17 g horizontally and 7.4 g vertically. A series of single frequency tests were performed in the two horizontal and one vertical direction at closely spaced frequencies (2 Hz) in the seismic range. Input acceleration was at the table limit for the lower frequencies and the malfunction limit for the higher frequencies. The accelerations required for the mounting location were 5.6 g (f/b), 5.1 g (s/s), and 1.6 g (v), which were enveloped by the test accelerations. Seismic qualification with substantial margin was demonstrated.

Based on inspection of the field installation, review of the qualification documents, and the applicant's response to questions, the benchboard is adequately qualified for the prescribed loads.

2.9 FUEL POOL RETURN LINE FLOW CONTROL VALVE (NSSS-9)

The fuel pool return line flow control valve is a 10 in. 150 lb class butterfly valve. It was supplied by Contromatics Corp. with model No. C-W2566-CC. Its actuator was supplied by Limitorque with model No. SMB/4BC. The valve and actuator are line mounted in the fuel pool return line at elevation 599 ft of the intermediate building.

The control valve and actuator were purchased to design specification SP-524-4549-00. Qualification of the valve and actuator for dynamic loading was accomplished by testing performed by Acton Environmental Testing Co., documented in report No. 12875-1, dated April 27, 1977. The valve and actuator assembly were rigidly attached to the test table at 45 degrees to the table input motion. The side to side natural frequency was determined to be 22.8 Hz. The vertical and front to back natural frequencies were determined to be above 33 Hz. The test specimen was subjected to sine beat testing with peak input acceleration levels of 3.0 g. Each test had 5 beats with 10 cycles per beat.

The tests were performed at the valve natural frequency and at the following frequencies: 1, 5, 10, 15, 20, 25, 30, 33 Hz. The tests were performed in four orientations to provide both in-phase and out-of-phase motion for each biaxial combination. The valve was actuated during each test and was observed to operate without malfunction. No valve leakage was observed during or after tests with the valve pressurized to 165 psi. The valve acceleration levels determined from the piping seismic analysis are s/s 1.014 g, f/b 0.522 g, and v 0.307 g.

Based on our observation of the field installation, review of the qualification reports and the responses by the applicant to our questions, the flow control valve is adequately qualified for the prescribed loads.

2.10 DIFFERENTIAL PRESSURE TRANSMITTER (NSSS-10)

The differential pressure transmitter (model No. 1151) was supplied by Rosemount to purchase part drawing No. 163C1563. This locally mounted item is in the auxiliary building at the 568 ft 4 in. elevation. It weighs 11.9 lb. The mounting consists of horizontal and vertical plates, with the equipment bolted with four 1/4 in. bolts. Seismic loads were considered in the qualification. Details of the qualification were in the report: H22 Local Panels Qualification Report, GE DRF No. A00-794-10, dated March 18, 1980.

This pressure transmitter measures water pressure in the residual heat removal shutdown cooling suction piping. It provides an electrical output signal that is proportional to the measured pressure. If the pressure rises to a predetermined high value, it provides a trip signal which will sound a control room annunciator alarm and illuminate an associated window message. During normal operation, this alarm is an indication of leakage (through closed isolation valves) from the reactor into the RHR shutdown cooling suction piping.

Qualification is based on tests performed on a similar unit. The test consisted of independent biaxial random input. In-phase and out-of-phase inputs were provided. The same set of tests were repeated after rotating the specimen 90 degrees about its vertical axis. There were five OBE and one SSE level tests performed. TRS were generated for each case. TRS, having a five percent damping, envelope the RRS with four percent damping satisfactorily. The ZPA for the TRS is higher than that required in each case. GE also stated the equipment has only a passive safety function and it did demonstrate pressure integrity.

Based on observation of the field installation, review of the qualification document and the answers provided by the applicant to our questions, this pressure transmitter is adequately qualified for the Perry site.

3.1 CONTROL COMPLEX CHILLED WATER PUMP (BOP-1)

The control complex chilled water pump is located at elevation 574 ft of the control complex building. It is a horizontal pump rated for 1600 gpm at 1750 rpm. The pump was supplied by the Ingersoll-Rand Co. The pump has serial No. 0378140. The pump motor is a 100 horsepower Westinghouse motor, catalog No. 7901-01-001. The pump and motor are both bolted to a common skid with four 1 in. diameter bolts. The skid is in turn bolted to the floor with six 7/8 in. diameter embedded bolts.

Qualification of the pump and motor for seismic loading was performed by analysis. Both were purchased to design specification SP-750-4549-00. The structural integrity and operability analysis of the pump was performed by the Ingersoll-Rand Company, documented in report No. 94Q-211-2, dated November 23, 1982. The motor analysis was performed by Westinghouse, documented in report No. AL50037, dated June 16, 1983. The pump analysis was performed considering static seismic loading in combination with normal operating loads per the requirements of ASME Boiler and Pressure Vessel Code Section III, Subsection ND. Natural frequencies of the pump were determined to be greater 33 Hz in all three directions. Static loading coefficients equal to the seismic ZPA were used for the seismic loading portions of the analysis. These values were 0.09 g in all three directions for OBE and 0.18 g for SSE loading. All stresses for combined seismic and normal operating loads were within the ASME Section III, Subsection ND allowables. The pump motor analysis was also performed using ZPA static coefficient loading. The stresses from this analysis were also within allowable stresses and the motor rotor deflections were less than the provided clearance, assuring operability of the motor.

Based on our observation of the field installation, review of the qualification reports and the responses by the applicant to our questions, the chilled water pump and motor is adequately qualified for the defined seismic loads.

3.2 LEAD-ACID ELECTRICAL STORAGE BATTERIES AND RACKS (BOP-2)

The batteries (model No. 2GN-15) and battery racks were supplied by Exide Power Systems Division of ESB Corporation. This 125 Vdc system, measuring 108 W x 38.25 H x 48.75 D in., is located in the control complex at the 638.5 ft elevation. The battery cells are clear plastic, mounted on an open steel lattice rack. The rack is a three bay, two step type which is welded to the floor. There are sixteen 2.5 in. long welds per rack. The purchase specification is contained in the report: Design, Fabrication, and Delivery of Class 1E Station Batteries and Racks, SP-554-4549-00, Rev. 1, dated January 17, 1978. The qualification of the batteries is documented in the report: Nuclear Environmental Qualification Program on Type GN Lead Acid Electrical Storage Batteries, No. 45001-1, rev. A, dated November 30, 1981, done by Wyle Laboratories. The test was performed on a 12 battery configuration rack. Through analysis the 15 battery configuration was found to be adequate. This rack analysis is contained in the report titled: Comparison Test and Analysis of Two Step "G" Size, 3 Bay High Seismic Battery Rack, No. A-3-82 dated February 10, 1982. These reports were reviewed by Gilbert/Commonwealth.

The qualification of the batteries and racks was done through test. Required peak acceleration for the location were:

	s/s	f/b	v
OBE	2.53 g	2.53 g	2.53 g
SSE	3.2 g	3.2 g	3.2 g

In the laboratory configuration, a three bay, two step 12 battery arrangement was welded to the test table. There were two types of tests performed, resonance search and qualification tests. Resonance searches were performed in the range of 1 to 200 Hz. The following fundamental frequencies were detected (for the racks):

s/s = 11.5 Hz, f/b = 15.2 Hz, and v = 35 Hz.

Qualification tests were independent, biaxial and random. In-phase and out-of-phase inputs were provided. The specimen was subjected to the same set of tests after a 90 degree rotation about the vertical axis. The peak acceleration for the tests were:

	s/s	f/b	v
OBE	6.25 g	6.25 g	6.25 g
SSE	10.0 g	9.8 g	7.4 g

Ten OBE and two SSE level tests were performed. These tests were performed after thermal and radiation aging equivalent to 20 years of normal operation. TRS were generated for each test. TRS enveloped the RRS for each case for 2% damping.

There was no anomaly detected during the tests. Subsequently, an analysis was performed to extend the adequacy of the rack to a 15 battery configuration. The fundamental frequencies of the model were:

s/s = 11.17 Hz, f/b = 15.98 Hz and v = 34.51 Hz.

The critical stresses were:

Element	Total Stress (psi)	Allowable Stress (psi)
support angle	950	28,000
lower left frame support	27,795	31,110

Allowable stresses were taken from AISC Sections 1.5 and 1.6.

Based on the observation of the field installation (with spacer), review of the qualification documents and the answers provided by the applicant to our questions the batteries and racks are adequately qualified for 20 years service in the seismic site environment.

3.3 FOUR INCH GLOBE VALVE (BOP-3)

The globe valve (MPL No. 11M51F0010B, model No. 81300) was supplied to design specification No. SP-521-02 by Borg Warner. It is located in the reactor building at an elevation of 670 ft. The valve weighs 423 lb (with actuator), and has dimensions of 15 x 22 x 40 in. It is supported by the piping to which it is welded. The valve is part of the combustible gas control system, where it provides isolation when shut and throttling when open.

The valve was qualified by analysis to the ASME Code (Borg Warner report No. NSR 81300, revision C, Seismic Analysis of 4 in.-300 lb Carbon Steel, Motor Operated Globe Valve, April 7, 1982). The fundamental natural frequency of the valve was established at 40.57 Hz by finite element analysis. The valve was then qualified using an equivalent static analysis. A conservative, simply supported boundary condition was used. Accelerations taken from the associated piping analysis were applied. All stresses were below the allowables. The maximum critical deflection (0.0142 in.) was less than the machining tolerances.

Based on our observation of the field installation and review of the qualification documents, the 4 in. globe valve is adequately qualified for the prescribed loads.

3.4 SOLENOID VALVE (BOP-4)

The solenoid valve (MPL No. 1D17F0079A, model No. 77JJ-004) was supplied to design specification No. SP-597-4549-00 by Target Rock Corp. It is located in the intermediate building at an elevation of 635 ft. The valve weighs 30 lb and is 15 in. long by 6 in. diameter. It is supported at the base by an integral attachment plate welded to a beam cantilevered from the wall. The valve is part of the plant radiation monitoring system, where it serves a containment isolation function.

The valve was qualified by analysis (Target Rock report No. 3215, Seismic Analysis of TRC Models 77JJ-001 through -004, February 18, 1982). The fundamental natural frequency was established at 117 Hz by a beam type calculation. The valve was qualified with a static analysis of an equivalent cantilevered beam. A 3 g acceleration was applied in all three orthogonal directions. This is well above the ZPA value for the mounting location. The maximum stress was calculated to be 7756 psi, vs. an allowable of 18750 psi. Operability was assured by a calculated maximum critical deflection of 0.0007 in. vs. a 0.005 in. allowable.

The design specification made reference to IEEE 344-1975 without reference to Regulatory Guide (R.G.) 1.100. Although a check of the analysis showed no concern for this particular item, there was a concern that the modifications of IEEE 344-1975 stated in R.G. 1.100 were not generically applied. However, the commitments made by the applicant in the FSAR (in tables 1.8-1, 8.1-2, and in Section 3.10.1.1.3.4) indicate that the R.G. 1.100 modification to IEEE 344-1975 are part of the generic requirements. This was supported by the fact that this problem was not encountered with any other SQRT item.

During field inspection of this valve, a nearby valve (No. 1D17F071A) was observed to be questionably supported. Lateral motion of the valve actuator was restrained by a cantilevered support (No. 1H51P071) which was not positively attached to the actuator. Contact between actuator and valve was through nonparallel surfaces. This could have resulted in an interaction where the contact load would force the support to slip up, allowing the valve to move laterally. However, hand calculations showed that the support was sufficiently stiff in the vertical direction to prevent such motion.

Based on our observation of the field installation, review of the qualification documents, and the applicant's answers to our questions, the solenoid valve is adequately qualified for the prescribed loads.

3.5 EMERGENCY SERVICE WATER TRAVELING WATER SCREEN CONTROL PANEL (BOP-5)

The emergency service water traveling water screen control panel, supplied by Rexnard Control Products, is located in the emergency service water pumphouse at elevation 586 ft 6 in. This control box is custom made. It supports electrical devices required to provide control functions to the emergency service water traveling water screen. These functions must be provided for both hot standby and cold shutdown conditions in addition to post-accident and after normal shutdown conditions. The control box is pedestal mounted. The base of the pedestal is bolted to the pumphouse floor with four 1/2 in. diameter embedded bolts.

The qualification of the control box and associated electrical devices was performed by a combination of analysis and testing. A response spectra analysis of the pedestal and box was performed by LeRoy A. Lutz Computerized Structural Design, Inc. Documentation of this analysis is contained in Lutz report No. 78321, dated May 31, 1979. The pedestal natural frequencies were determined to be 9.73 Hz s/s, 11.91 Hz f/b, and 20.44 Hz vertical. The results of the analysis demonstrate that structural integrity of the control box will be maintained for postulated seismic loading. An additional analysis was performed by Gilbert/Commonwealth to determine the in cabinet response spectra for the associated electrical devices. This analysis is documented by Gilbert report No. 94Q-364-1-0, dated October 5, 1983. Since the associated electrical devices are used in several applications throughout the Perry plant, they were qualified separately by seismic qualification tests with test response spectra which sufficiently envelopes the control box in-cabinet required response spectra.

Based on our observation of the field installation, review of the qualification reports and the responses by the applicant to our questions, the control panel is adequately qualified for the defined seismic loads.

3.6 MOTOR CONTROL CENTER (BOP-6)

The motor control centers, supplied by Eaton/Cutler-Hammer (model No. F245), are located at elevation 586 ft 6 in. of the emergency service water pumphouse. This equipment consists of electrical switchgear mounted in cabinets which are 96 in. long, 24 in. wide, 90 in. high with a total weight of 2240 lb. The cabinet base is welded (front and back) to embedded angle iron with intermittent 1/4 in. fillet welds 3 in. long, spaced at six inches.

This item was purchased to design specification SP-750-4549-00. The cabinet and its contents were qualified for seismic loads by testing. The report documenting the seismic qualification of this item was prepared by Patel Engineering, report No. PEI-TR-83-9, dated March 24, 1983. The testing was performed by Farwell and Hendricks, documented by report No. 10049, dated March 18, 1983. The cabinet was welded to a test fixture and subjected to low level resonance search and biaxial random motion seismic tests. The cabinets natural frequencies were determined to be 17.8 hz side to side, 18.9 hz front to back, and greater than 33 hz vertically. The seismic tests consisted of five OBE and one SSE biaxial tests in each test direction; front to back/vertical and side to side/vertical. The test response spectra enveloped the required response spectra for both OBE and SSE tests. Functional operability of the electrical devices contained in the cabinet was verified during and after the tests. A seperate analysis was performed to account for the effects of a top entry cable. This analysis is documented by GAI calculation No. 40.01 dated June 8, 1983. It demonstrated that stresses for a malleable iron connector would be within the allowable stress.

Based on our observation of the field installation, review of the qualification reports and the responses by the applicant to our questions, the motor control center is adequately qualified for the defined seismic loads.

3.7 VERTICAL PUMP (BOP-7)

The vertical pump (MPL No. 1P45C0002, model No. VIT) was supplied by Gould Pumps to design specification Nos. SP-750-4549-00, Revision 1, and SP-501-4549-00, Revision 2. It is located in the emergency service water pumphouse at an elevation of 586.5 ft. The pump weighs 3730 lb (flooded) and is 41.75 ft long by 11-3/8 in. diameter. The pump is attached to the floor with four 1-3/8 in. bolts. Seismic lateral supports are provided for the unit. The pump is part of the emergency service water system, where it supplies water for component cooling.

The pump was qualified by analysis. (McDonald Engineering Analysis Company report No. ME-453, Seismic Analysis of Vertical Pump, September 16, 1977). The fundamental frequency of the pump was established at 13.6 Hz by a 2-D finite element analysis. The computer code used, ICES-STRUDL, is a well known and widely used code. Response spectrum analysis indicated that the most strongly challenged area of the pump was the discharge head base plate. The calculated stress there was 26,155 psi, vs. an allowable of 26,250 psi. Operability was ensured by a maximum critical deflection of 0.0010 between rotor and starter. The maximum allowable deflection in this area is 0.0040 in.

Based on our observation of the field installation and review of the qualification documents, the vertical pump is adequately qualified for the prescribed loads.

3.8 AIR HANDLING UNIT (BOP-8)

The air handling unit is located at elevation 574 ft of the auxiliary building. It is a floor mounted, box shaped housing with an 11 in. diameter fan. The fan motor is mounted external to the fan housing. The total weight of this unit is approximately 700 lb. The unit was supplied by Carrier Air-Conditioning Company with model No. 3913A050. The fan motor was supplied by Reliance Electric Company. The unit is floor mounted with four 1/2 in. diameter bolts. This unit supplies ventilation and cooling air to the AB-3 area of the auxiliary building, and is required during both hot standby and cold shutdown conditions.

This unit was purchased to design specification SP-750-4549-00. Seismic qualification of the air handling unit was performed by John Henry Associates, Inc. documented by Report No. JHA-76-73A, dated December 29, 1978. This item was qualified by response spectrum analysis. A three dimensional computer model comprised of beam and plate elements was used in the analysis of the air handling unit. The computer code STARDYNE was used to perform this analysis. Stresses from combined seismic and normal loading, including a pressure loading of 5 in. of water, were evaluated. They were compared to the requirements of the ASME Boiler and Pressure Vessel Code Section III Subsection NF. All stresses were determined to be within the specified allowables. The seismic loading was derived from the auxiliary building floor response spectra for the 599 ft elevation with 4% OBE and 7% SSE damping. The unit natural frequencies were determined to be 28.8 Hz (s/s), 14.6 Hz (f/b), and 6.1 Hz (v). The relative displacement between the fan and housing was determined to be 0.0039 in. compared to a clearance of 0.25 in., which assures operability of the unit during a postulated seismic event.

The fan motor was qualified by Reliance Electric, documented by report No. 77-A-34, dated June 29, 1977. The motor was analyzed for seismic loading which accounted for the flexibility of its mounting location on the fan housing. The motor stresses were determined to be below allowable values and the motor rotor to stator relative displacements were one third the provided clearance.

Based on our observation of the field installation, review of the qualification reports and the responses by the applicant to our questions, the air handling unit is adequately qualified for the defined seismic loads.

3.9 POWER SUPPLY PANEL (BOP-9)

The power supply panel (MPL No. 1M51S0002, no model No.) was supplied by Westinghouse Corp to design specification Nos. SP-750-4549-00, and SP-628-4549-00. It is located in the control building at an elevation of 620.5 ft. It is attached to the floor with eight 3/4 in. bolts. The panel weighs 950 lb and has dimensions of 61 L x 34.2 W x 23.5 h in. It is part of the combustible gas control system, where it supplies power to a hydrogen recombiner.

The panel was qualified by test (Westinghouse report Nos. WCAP-7709L supplements 1 through 7, Electric Hydrogen Recombiner for Water Reactor Containment). Two series of tests were performed. Attachment to the test table for both was with eight 1/2 in. cap screws. The first test was a resonance search (fundamental natural frequency of 3.5 Hz). This was followed by eleven sine beat tests distributed in the 1.25 to 33.5 Hz range. Testing was to 8 g input accelerations in two horizontal directions in combination with a 5.3 g vertical. It was performed in four orientations on the table. A later test series on the cabinet consisted of five multifrequency tests to a generic OBE RRS that enveloped the Perry RRS. Each test showed the equipment to be in operating condition.

A wiring harness strap was added and a temperature indicator mounting bracket was modified early in the sine beat testing. Four consecutive sine beat tests were successfully run following the modification. The panel installed in the field reflected the modifications.

During the field inspection, the panel was observed to be located next to a sheetrock wall. There was a concern about the seismic integrity of the wall. Followup indicated that seismic loads were considered in the design of the wall.

Based on our inspection of the field installation, review of the qualification documents, and the applicant's responses to questions, the power supply panel is adequately qualified for the prescribed loads.

3.10 LEVEL SWITCH (BOP-10)

The level switch (model No. 580A-1) was supplied by ITT Barton according to the purchase Specification No. SP-598. It has a range of 0-30 in. of water. This equipment, weighing about 17 lb, was located in the intermediate building at the 620 ft elevation. It was mounted on a post attached to the floor with four 5/16 in. bolts. Seismic loads were considered in the qualification. Details of the qualification are in the document: ITT Barton Qualification Report, Nos. R3-580A-9 and 94Q-689-2-0 (TABC) as supplemental volumes 1, 2, 3, and 4 dated December 23, 1983. They were reviewed by Gilbert/Commonwealth.

This auxiliary mixing tank level indicator, part of the standby liquid control system, was qualified based on tests performed on a structurally similar unit (Model No. 580A-0). The mounting for the test specimen was similar to the field mounting. Input to the table for the tests was independent, biaxial and random. The inputs were in-phase and out-of-phase. The same set of tests was repeated for another orientation (rotated 90 degrees about the vertical axis from the first orientation). There were five OBE and one SSE level tests performed. The input duration in each case was 30 seconds. TRS were generated for each test and compared to the RRS for 2% damping. Envelopment in every case (including the ZPA) is satisfactory.

Two anomalies were observed during the qualification process. Their resolution follows:

1. The acceptance criteria according to the purchase specification SP-598 was ±4.0%. The test indicated an accuracy of ±9%. The applicant stated the following in response to our question:

   "The SP-598 ±4% accuracy requirement is the catalog accuracy. System analysis in Section 6 of the qualification package demonstrates that ±9% is adequate for the switch application/function."

2. There was contact chatter detected during the seismic tests. In response to our question, the applicant submitted the following:

"Section 8 of the lab report states that the switch was functional before, during and after each seismic test and the contact chatter did not effect the switch operation at its set point. Therefore, it does not constitute a failure."

Based upon the observation of the field installation, review of the qualification report and especially the applicant's responses to questions, the level switch is adequately qualified for the seismic environment of the Perry site.

3.11 MEDIUM VOLTAGE MODULAR ELECTRICAL PENETRATION (BOP-11)

The medium voltage modular electrical penetration is located at elevation 659 ft of the containment. This component penetrates through the containment wall. It must provide electrical service while maintaining the containment pressure boundary during both hot standby and cold shutdown conditions. The penetration is 18 in. in diameter by 128 in. long and weighs approximately 1200 lb. It was supplied by Westinghouse with model No. WX 33328.

This item was qualified for combined seismic and hydrodynamic loading through testing performed by Westinghouse. This testing is documented by report No. PEN-TR-82-52, dated December 1, 1982. The design specification for seismic qualification to which the medium voltage penetrations were purchased is SP-750-4549-000. A prototype penetration was welded to a rigid test table nozzle for low level resonance search and biaxial dynamic testing. Its natural frequencies were determined to be 11 Hz, 15 Hz, and 22 Hz. The dynamic tests were pseudo-biaxial.

The test specimen was mounted at 45 degrees to the test motion. The specimen was rotated 90 degrees between tests to account for in-phase and out-of-phase effects. The specimen was subjected to 20 OBE level and 4 SSE level tests. The test motion was random, multifrequency. The required response spectra peaks for elevation 688 ft. 6 in. of the containment are as follows:

	s/s	f/b	v
OBE	3.23 g	3.23 g	2.75 g
SSE	4.5 g	4.5 g	3.35 g

The TRS fully enveloped the RRS with spectral peaks as follows:

	s/s	f/b	v
OBE	8.2 g	8.2 g	8.2 g
SSE	10.5 g	10.5 g	10.5 g

The penetration performed its electrical function and maintained its leak tight function during and after the dynamic tests.

Based on our observation of the field installation, review of the qualification reports and the responses by the applicant to our questions, the medium voltage modular electrical penetration is adequately qualified for the defined seismic loads.

3.12 CONTROL ROOM CEILING

The control room ceiling was chosen for review, for seismic loads adequacy, during the course of the audit. This allowed the applicant insufficient time to locate and provide the documentation during the audit. The documentation was subsequently provided. It demonstrated that the seismic loads were adequately considered in the design and installation of the control room ceiling.

4. FINDINGS AND CONCLUSION

The review of the Perry plant will be completed when the following open items are closed:

1. The applicant must supply confirmation that all safety related equipment has been fully qualified. This requirement may be waived for a limited number of items, provided that justifications for interim operation have been submitted and approved for all unqualified safety related equipment prior to the fuel load.

2. New hydrodynamic loads (related to LOCA) have been calculated and approved by the NRC. The impact of the new loads on the qualification of equipment must be assessed. A schedule for the assessment and confirmation that the affected equipment, if any, has been requalified under the new loads, is needed.

3. The applicant must report whether or not the rod position multiplexer cabinet (NSSS-2) is safety-related.

Based on our review, we conclude that, pending resolution of all open items, an appropriate qualification program has been defined and implemented for the seismic Category I mechanical and electrical equipment which will provide reasonable assurance that such equipment will function properly during and after the excitation due to the vibratory forces imposed by a safe shutdown earthquake in combination with hydrodynamic and normal operating loads.

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TABLE 1. LIST OF ATTENDEES

Name	Company
N. Anderson	GE
G. Bagchi	NRC/DE
J. Boseman	GE
B. S. Ferrell	CEI
B. Fleming	GE
M. Gaballa	GAI
D. R. Green	CEI
D. Hardy	GE
J. Ionnidi	GAI
J. Kelso	GE
C. Kido	EG&G Idaho
G. Koenig	GAI
R. Kowicki	CEI
M. R. Kritzer	CEI
S. Litchfield	CEI
D. P. Lohisey	CEI
S. R. Mannon	GAI
C. F. Miller	EG&G Idaho
H. Patel	Bechtel
H. A. Putre	CEI
T. Rockwell	GAI
N. Romney	DE/EQB
F. C. Rosch	GAI
M. J. Russell	EG&G Idaho
D. Shamis	GE
J. N. Singh	EG&G Idaho
J. Smith	CEI
F. R. Stead	CEI
J. J. Stefano	NRC/DL
J. Sunden	CEI
R. A. Svotelis	CEI
E. B. Thomas	CEI
T. J. Thompson	CEI
L. E. Wise	GAI

APPENDIX J

CONFORMANCE TO REGULATORY GUIDE 1.97
PERRY NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

CONFORMANCE TO REGULATORY GUIDE 1.97
PERRY NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

A. C. Udy

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Idaho Falls, Idaho 83415

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ABSTRACT

This EG&G Idaho, Inc., report provides a review of the submittals for Regulatory Guide 1.97, Revision 2, for the Perry Nuclear Power Plant, Unit Nos. 1 and 2. Any exception to the guidelines of Regulatory Guide 1.97 are evaluated and those areas where sufficient basis for acceptability is not provided are also identified.

FOREWORD

This report is supplied as part of the "Program for Evaluating Licensee/ Applicant Conformance to RG 1.97," being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Systems Integration, by EG&G Idaho, Inc., NRC Licensing Support Section.

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Docket Nos. 50-440 and 50-441

CONTENTS

CONFORMANCE TO REGULATORY GUIDE 1.97
PERRY NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

1. INTRODUCTION

On December 17, 1982, Generic Letter No. 82-33 (Reference 1) was issued by D. G. Eisenhut, Director of the Division of Licensing, Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating licenses and holders of construction permits. This letter included additional clarification regarding Regulatory Guide 1.97, Revision 2 (Reference 2), relating to the requirements for emergency response capability. These requirements have been published as Supplement 1 to NUREG-0737, "TMI Action Plan Requirements" (Reference 3).

The Cleveland Electric Illuminating Company, applicant for the Perry Nuclear Power Plant, provided a response to the generic letter on April 15, 1983 (Reference 4). Section 6.2 of the generic letter is addressed in Amendment 12 to the Final Safety Analysis Report (FSAR), dated July 19, 1983 (Reference 5).

This report provides an evaluation of this material.

2. REVIEW REQUIREMENTS

Section 6.2 of NUREG-0737, Supplement 1, sets forth the documentation to be submitted in a report to the NRC describing how the applicant meets the Guidance of Regulatory Guide 1.97 as applied to emergency response facilities. The submittal should include documentation that provides the following information for each variable shown in the applicable table of Regulatory Guide 1.97.

1. Instrument range
2. Environmental qualification
3. Seismic qualification
4. Quality assurance
5. Redundance and sensor location
6. Power supply
7. Location of display
8. Schedule of installation or upgrade.

Further, the submittal should identify deviations from the guidance in the regulatory guide and provide supporting justification or alternatives.

Subsequent to the issuance of the generic letter, the NRC held regional meetings in February and March 1983, to answer licensee and applicant questions and concerns regarding the NRC policy on this matter. At these meetings, it was noted that the NRC review would only address exceptions taken to the guidance of Regulatory Guide 1.97. Further, where licensees or applicants explicitly state that instrument systems conform to the provisions of the guide it was noted that no further staff review would be necessary.

Therefore, this report only addresses exceptions to the guidance of Regulatory Guide 1.97. The following evaluation is an audit of the applicant's submittals based on the review policy described in the NRC regional meetings.

3. EVALUATION

The applicant provided a response to Item 6.2 of the NRC generic letter 82-33, on July 19, 1983. The response, Table 7.1-4 of Amendment 12 to the FSAR, describes the applicant's position on post-accident monitoring instrumentation. This evaluation is based on that material.

3.1 Adherence to Regulatory Guide 1.97

The applicant states that "the variables and associated requirements of NRC Regulatory Guide 1.97, Rev. 2. have been reviewed and utilized as guidance in CEI's plans to provide instrumentation to assess plant and environs conditions during and following an accident."

Therefore, it is concluded that the applicant has provided an explicit commitment on conformance to the guidance of Regulatory Guide 1.97, except for those deviations that were identified by the applicant as noted in Section 3.3.

3.2 Type A Variables

Regulatory Guide 1.97 does not specifically identify Type A variables, i.e., those variables that provide information required for operator controlled safety actions. The applicant classified the following instrumentation channels as Type A variables.

1. Neutron flux

2. Coolant level in reactor

3. Reactor coolant system (RCS) pressure

4. Drywell pressure

5. Primary containment pressure

6. Suppression pool water level

7. Containment and drywell hydrogen concentration

8. Suppression pool water temperature

9. Drywell atmosphere temperature

10. Containment atmosphere temperature

All of the above variables, except containment atmosphere temperature are also type B, C, or D variables. All meet the Category 1 requirements consistent with the requirements for Type A variables, except for neutron flux. This is addressed in Section 3.3.1.

3.3 Exceptions to Regulatory Guide 1.97

The applicant identified the following deviations from the recommendations of Regulatory Guide 1.97.

3.3.1 Neutron Flux

Regulatory Guide 1.97 recommends Category 1 instrumentation for this variable. The applicant's instrumentation is not Category 1. The applicant states that it is installed in accordance with the Category 2 requirements. The applicant has not provided justification for this deviation.

In the process of our review of neutron flux instrumentation, we note that the mechanical drives of the detectors have not satisfied the environmental qualification requirement of Regulatory Guide 1.97. This deviation is similar to most BWRs. A Category 1 system that meets all the criteria of Regulatory Guide 1.97 is an industry development item. Based on our review, we conclude that the existing instrumentation is acceptable for interim operation. The licensee should follow industry development of this equipment, evaluate newly developed equipment, and install Category 1 instrumentation when it becomes available.

3.3.2 Drywell Sump Level
Drywell Drain Sumps Level

Regulatory Guide 1.97 recommends Category 1 instrumentation for these variables. The applicant has supplied Category 3 instrumentation for the sump leakage flow rate instead of sump level. The drywell sump systems are automatically isolated at the primary containment penetration should an accident signal occur.

We find the sump level detection is a method of determining leakage from the reactor coolant system that is specified in Regulatory Guide 1.45. This leakage through the sump drain is measured.

For small leaks, this Category 3 instrumentation will continue to function as the drywell temperature and pressure will not have changed significantly. Therefore, the sump drains flow can be used as a leading indicator of reactor coolant system leakage. For larger leaks, the sumps will fill promptly, negating this instrumentation because the sump drain lines isolate due to the increase in drywell pressure caused by the accident. The sumps can be assumed full once containment isolation occurs at 2 psig.

In either case, we find the Category 3 instruments provided for this variable acceptable.

3.3.3 Radiation Level in Circulating Primary Coolant

A continuous, direct measurement of this variable is not provided. The applicant indicates that radiation level measurements to indicate fuel cladding failure are provided by the following:

1. Condenser off-gas radiation monitors

2. Main steamline radiation monitors

3. Post-accident sampling system.

Based on the justification provided by the applicant, we conclude that the instrumentation supplied for this variable is adequate, and therefore, acceptable.

3.3.4 Suppression Chamber Spray Flow

Regulatory Guide 1.97 specifies Category 2 instrumentation for this variable with a range from 0 to 110 percent of design flow. The purpose of this instrumentation is to monitor the operation of this primary containment-related system.

The applicant has not provided information on instrumentation for this variable, nor shown it to be in conformance with the recommendations of Regulatory Guide 1.97. The applicant should provide this information.

3.3.5 Main Steamline Isolation Valves (MSIV) Leakage Control System Pressure

The applicant has provided actual MSIV leakage flow (0 to 100 SCFH) instrumentation instead of the recommended pressure instrumentation.

The applicant considers monitoring the leakage flow as less ambiguous than the indication recommended by the regulatory guide. We find this deviation from the recommendation of Regulatory Guide 1.97 acceptable.

3.3.6 Standby Liquid Control System (SLCS) Flow

The applicant has elected not to implement this variable as recommended in Regulatory Guide 1.97. The justification given by the applicant is that the SLCS pump discharge pressure provides indication that the SLCS pump is operating and that the level indication in the SLCS storage tank gives indication that flow is occurring.

We find that the above indications are valid for an alternative SLCS flow indication.

3.3.7 RHR Heat Exchanger Outlet Temperature

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable with a range from 32 to 350°F. The applicant has chosen not to monitor this variable directly, but to use the RHR service water flow (used to cool the RHR water) instead to verify system operation. They also monitor the heat exchanger bypass valve position in the control room. The valve is used to bypass a portion of the RHR water around the heat exchanger to regulate the RHR water temperature.

As the heat exchanger bypass valve is used to regulate the RHR system water temperature, we conclude that the applicant has instrumentation for this variable. This information has not been supplied in accordance with Section 6.2 of Reference 3. The applicant should provide additional information on the instrumentation supplied for, or related to, this variable.

3.3.8 Cooling Water Temperature to ESF System Components

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable with a range of 32 to 200°F. The applicant provides instrumentation for the emergency closed cooling loop temperature with a range of 50 to 150°F, and for the emergency service water loop inlet temperature with a range of 0 to 100°F.

Sections 9.2.1 and 9.2.2 of the FSAR describe these systems. The emergency service water system is an open loop system, with water taken from Lake Erie. The expected temperature is between 32 and 80°F. Therefore, the range of 0 to 100°F is acceptable. The emergency closed cooling loop water is cooled by the emergency service water in heat exchangers. The FSAR indicates that the maximum expected temperature is 95°F. Therefore, the upper limit of the range (150°F) is acceptable. We could find no basis for accepting the lower limit of 50°F.

The applicant should provide justification for the lower limit of the range for the emergency closed cooling water temperature being 50°F.

4. CONCLUSIONS

Based on our review, we find that the applicant either conforms to, or is justified in deviating from, the guidance of Regulatory Guide 1.97 with the following exceptions:

1. Neutron flux--the applicant's present instrumentation is acceptable on an interim basis until Category 1 instrumentation is developed and installed (Section 3.3.1).

2. Suppression chamber spray flow--the applicant should show the instrumentation for this variable is in conformance with the recommendations of the regulatory guide (Section 3.3.4).

3. RHR heat exchanger outlet temperature--the applicant should provide additional information on the instrumentation for this variable (Section 3.3.7).

4. Cooling water temperature to ESF system components--the applicant should provide justification for the lower limit of the range for the emergency closed cooling water temperature instrumentation being 50°F (Section 3.3.8).

5. REFERENCES

1. NRC letter, D. G. Eisenhut to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement No. 1 to NUREG-0737--Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

2. Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident, Regulatory Guide 1.97, Revision 2, U.S. Nuclear Regulatory Commission (NRC), Office of Standards Development, December 1980.

3. Clarification of TMI Action Plan Requirements, Requirements for Emergency Response Capability, NUREG-0737 Supplement No. 1, NRC, Office of Nuclear Reactor Regulation, January 1983.

4. Cleveland Electric Illuminating Company (CEI) letter, M. R. Edelman to B. J. Youngblood, NRC, "Our Response to Generic Letter 82-33, NUREG-0737, Supplement 1," April 15, 1983, PY-CEI/NRR-0032 L.

5. CEI letter, M. R. Edelman to H. R. Denton, NRC, "Amendment 12," July 19, 1983, PY-CEI/NRR-0059 L.

APPENDIX K

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2
(PHASE I)

EGG-HS-6264

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2
(PHASE I)

Docket Nos. 50-440 and 50-441

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ABSTRACT

The Nuclear Regulatory Commission (NRC) has requested that all nuclear plants, either operating or under construction, submit a response of compliancy with NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." EG&G Idaho, Inc., has contracted with the NRC to evaluate the responses of those plants presently under construction. This final report is a result of EG&G's review of the responses submitted for the Perry Nuclear Power Plant, Units 1 and 2 to the requirements of Section 5.1.1 of NUREG-0612 (Phase I). Sections 5.1.2, 5.1.4, 5.1.5, and 5.1.6 (Phase II) will be covered in a separate report.

EXECUTIVE SUMMARY

Perry Nuclear Power Plant, Units 1 and 2 comply with the intent of the requirements of NUREG-0612.

CONTENTS

TABLES

CONTROL OF HEAVY LOADS AT NUCLEAR POWER PLANTS
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
PERRY NUCLEAR POWER PLANT, UNITS 1 AND 2
(PHASE I)

1. INTRODUCTION

1.1 Purpose of Review

This technical evaluation report documents the EG&G Idaho, Inc., review of general load-handling policy and procedures at The Cleveland Electric Illuminating Company (CEICO), Perry Nuclear Power Plant, Units 1 and 2. This evaluation was performed with the objective of assessing conformance to the general load-handling guidelines of NUREG-0612. "Control of Heavy Loads at Nuclear Power Plants" [1], Section 5.1.1.

1.2 Generic Background

Generic Technical Activity Task A-36 was established by the U.S. Nuclear Regulatory Commission (NRC) staff to systematically examine staff licensing criteria and the adequacy of measures in effect at operating nuclear power plants to assure the safe handling of heavy loads and to recommend necessary changes to these measures. This activity was initiated by a letter issued by the NRC staff on May 17, 1978 [2], to all power reactor applicants, requesting information concerning the control of heavy loads near spent fuel.

The results of Task A-36 were reported in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." The staff's conclusion from this evaluation was that existing measures to control the handling of heavy loads at operating plants, although providing protection from certain potential problems, do not adequately cover the major causes of load-handling accidents and should be upgraded.

In order to upgrade measures for the control of heavy loads, the staff developed a series of guidelines designed to achieve a two-phase objective using an accepted approach or protection philosophy. The first portion of the objective, achieved through a set of general guidelines identified in NUREG-0612, Article 5.1.1, is to ensure that all load-handling systems at nuclear power plants are designed and operated such that their probability of failure is uniformly small and appropriate for the critical tasks in which they are employed. The second portion of the staff's objective, achieved through guidelines identified in NUREG-0612, Articles 5.1.2 through 5.1.5, is to ensure that, for load-handling systems in areas where their failure might result in significant consequences, either (a) features are provided, in addition to those required for all load-handling systems, to ensure that the potential for a load drop is extremely small (e.g., a single-failure-proof crane) or (b) conservative evaluations of load-handling accidents indicate that the potential consequences of any load drop are acceptably small. Acceptability of accident consequences is quantified in NUREG-0612 into four accident analysis evaluation criteria.

The approach used to develop the staff guidelines for minimizing the potential for a load drop was based on defense in depth and is summarized as follows:

- Provide sufficient operator training, handling system design, load-handling instructions, and equipment inspection to assure reliable operation of the handling system

- Define safe load travel paths through procedures and operator training so that, to the extent practical, heavy loads are not carried over or near irradiated fuel or safe shutdown equipment

- Provide mechanical stops or electrical interlocks to prevent movement of heavy loads over irradiated fuel or in proximity to equipment associated with redundant shutdown paths.

Staff guidelines resulting from the foregoing are tabulated in Section 5 of NUREG-0612.

1.3 Plant-Specific Background

On December 22, 1980, the NRC issued a letter [3] to the Cleveland Electric Illuminating Company, the applicant for the Perry Nuclear Power Plant, Units 1 and 2, requesting that the applicant review provisions for handling and control of heavy loads at PNPP Units 1 and 2, evaluate these provisions with respect to the guidelines of NUREG-0612, and provide certain additional information to be used for an independent determination of conformance to these guidelines. On June 19, 1981, CEICo provided the initial response [4] to this request. Based on this information, a preliminary draft of this report was prepared and discussed with the applicant. Additional information was provided by the applicant in References [9, 10, 11]. The current (final) draft of this report was prepared from information contained in all these submittals.

2. EVALUATION AND RECOMMENDATIONS

2.1 Overview

The following sections summarize CEICo's review of heavy load-handling at Perry Nuclear Power Plant, Units 1 and 2 accompanied by EG&G's evaluation, conclusions, and recommendations to the applicant for bringing the facilities more completely into compliance with the intent of NUREG-0612. The applicant has indicated the weight of a heavy load for this facility (as defined in NUREG-0612, Article 1.2) as 1048 pounds.

2.2 Heavy Load Overhead Handling Systems

This section reviews the applicant's list of overhead handling systems which are subject to the criteria of NUREG-0612 and a review of the justification for excluding overhead handling systems from the above-mentioned list.

2.2.1 Scope

"Report the results of your review of plant arrangements to identify all overhead handling systems from which a load drop may result in damage to any system required for plant shutdown or decay heat removal (taking no credit for any interlocks, technical specifications, operating procedures, or detailed structural analysis) and justify the exclusion of any overhead handling system from your list by verifying that there is sufficient physical separation from any load-impact point and any safety-related component to permit a determination by inspection that no heavy load drop can result in damage to any system or component required for plant shutdown or decay heat removal."

A. Summary of Applicant's Statements

The applicant conducted a review of plant arrangements and provided tables and drawings in Reference 10 identifying cranes, hoists, or other overhead handling devices that have the capacity to lift more than 1048 pounds in buildings housing fuel or safe shutdown equipment. Separate tables were presented for each handling system. The tables included the impact area, loads and weights to be lifted, safety-related equipment, floor elevation, category for hazard elimination, and capacity of the handling system. Hazard elimination categories were defined and justification for the load classification was provided. The overhead handling systems used for the heavy load movements are identified in Table 2.1. All overhead handling systems excluded from further concern are listed on Table 2.2.

B. EG&G Evaluation

The applicant's response to the identification of overhead handling systems is very complete and well documented in Reference [10]. The drawings in this reference indicate that a notable amount of handling is accomplished by dollies, however, the applicant states that they will not be pulled by cranes [11]. EG&G is in agreement with the logic chart used as a basis for evaluation and analysis of the movement of heavy loads and commends CEICo's method of clear presentation. Tables 2.3, 2.4, and 2.5 identify weights, lift equipment, procedures, and analysis for loads that are to be handled by the Overhead Handling Systems identified in Table 2.1.

C. EG&G Conclusions and Recommendations

The Perry Nuclear Power Plants, Units 1 and 2 comply with the requirements of 2.2.1 above, of NUREG-0612 on Heavy Load Overhead Handling Systems.

TABLE 2.1. OVERHEAD HANDLING SYSTEMS USED FOR THE MOVEMENT OF HEAVY LOADS IN VICINITY OF SAFE SHUTDOWN EQUIPMENT PERRY, NUCLEAR POWER PLANT, UNITS 1 AND 2

Handling System	Capacity (lb)	Location
Emergency Service Water Pump House Crane	30,000	ESW Pump House
Reactor Building Crane	250,000	Reactor Building
Fuel-Handling Building Crane	250,000	Fuel-Handling Building

TABLE 2.2. OVERHEAD HANDLING DEVICES EXCLUDED FROM FURTHER CONCERN PERRY NUCLEAR POWER PLANT UNITS 1 and 2

Handling System	Capacity (lbs)	Location	*Hazard Elimination Category
Hoist 33-1, 33-2	30,000	Reactor Auxiliary Building	A
Hoist 34-1, 34-2	15,000	Reactor Auxiliary Building	A, B
Hoist 35-1, 35-2	30,000	Reactor Auxiliary Building	A, B
Hoist 42-A, B	1,048	Fuel Handling	NA
Hoist 43	8,000	Unit 2 Reactor Building	B, C
Hoist 44	70,000	Unit 1 Reactor Building	B, C
Hoist 45-1, 45-2	70,000	Reactor Building Annulus	B, C
Hoist 49-1, 49-2	6,000	Reactor Building	B, C
Hoist 50-1, 50-2	6,000	Reactor Building and Intermediate Building	B, C
Hoist 51-1, 52-2	10,000	Reactor Building	B, C
Hoist 58	8,000	Control Complex	A, B
Hoist 59	8,000	Control Complex	A, B
Hoist 60	8,000	Control Complex	A, B
Fuel Handling Machine	1,000	Fuel Handling Building	B
Hoist 36	12,000	Intermediate Building	B
Hoist 37-1, 37-2	30,000	Auxiliary Building	A, B
Hoist 38-1, 38-2	15,000	Auxiliary Building	A, B
Hoist 39-1, 39-2	30,000	Auxiliary Building	A, B
Hoist 40-1, 40-2	20,000	Auxiliary Building	A, B
Hoist 41-1, 41-2	20,000	Fuel Handling Building	B
Hoist 54-1A, B; 54-2A, B	2,000	Diesel Generator Building	B

TABLE 2.2. (continued)

Handling System	Capacity (lbs)	Location	*Hazard Elimination Category
Hoist 54-1C, D; 54-2C, D	2,000	Diesel Generator Building	B
Hoist 55-1A, 55-2A	40,000	Diesel Generator Building	B
Hoist 55-1B, 55-2B	40,000	Diesel Generator Building	A
Hoist 56-1A, B; 56-2A, B	2,000	Diesel Generator Building	B
Hoist 57-1, 57-2	20,000	Diesel Generator Building	B
Hoist 63	10,000	Intermediate Building	B
Hoist 53	30,000	Control Complex	B
Hoist 65-1A, B; 65-2A, B	4,500	Diesel Generator Building	A
Hoist 65-1C, D; 65-2C, D	4,500	Diesel Generator Building	A
Hoist 68-1A to H; 68-2A to H	1,000	Control Complex	NA
Hoist 64-1A to D, 64-2A to D	30,000	Steam Tunnel	A, C
Hoist 71	3,000	Fuel Handling	B
Hoist 72	1,000	Fuel Handling	NA
Hoist 75-1, 75-2	4,000	Reactor Building	A, B, C
Hoist 76-1, 76-2	4,000	Reactor Building	A, B, C
Hoist 46-1, 46-2	8,000	Reactor Annulus	A, B
Hoist 47-1, 47-2	8,000	Reactor Annulus	B
Hoist 52	6,000	Control Complex	B

* Hazard Elimination Categories.

A System Redundancy and Separation.

B No safety related equipment or critical piping in the load coverage path.

C Loads lifted only when reactor is in cold shutdown.

TABLE 2.3. EMERGENCY SERVICE WATER PUMP HOUSE CRANE--PNPP UNITS 1 AND 2

Load	Approximate Weight (lbs)	Lift Equipment	Procedure	Remarks
ESW Pumps and Equipment	30,000 (maximum)	--	MAP.1301	Procedures are necessary to ensure that any lifts of Train 'A' equipment will not be made over Train 'B' equipment and lifts of Train 'B' equipment will not be made over Train 'A.'

a. No special lift device identified by applicant.

TABLE 2.4. FUEL HANDLING BUILDING CRANE--PNPP UNITS 1 AND 2

Load	Approximate Weight (lbs)	Lift Equipment	Procedure	Remarks
Fuel Shipping Cask	250,000 (maximum)	N/A	MAP.1301	Crane coverage area is not over the spent fuel pool or any area where spent fuel or safety related equipment is housed. Analysis has shown that a load drop at the west end of the crane coverage area will not degrade the spent pool leakage integrity. Procedures and design limitations prevent the cask from being dropped more than 30 feet so that the impact design of the cask is not exceeded.

TABLE 2.5. REACTOR BUILDING CRANE--PNPP UNITS 1 AND 2

Load	Approximate Weight (lbs)	Lift Equipment	Procedure	Remarks
1. Dry Well Head	130,000	--	MAP.1301	Analysis has shown that drop over reactor would not cause damage to the vessel and internals, including fuel.
2. Vessel Head Piping Bundle	1,300	--	MAP.1301	Drop over reactor would not damage reactor and internals, including fuel.
3. Thermal Insulaiton RPV Top Head	10,000	--	MAP.1301	Drop over vessel would not cause damage to fuel and vessel.
4. PRV Head, O-rings and Head Strongback	236,000	Head Strongback	MAP.1301	Drop over vessel would not damage vessel and fuel.
5. Dryer-Separation Strongback	8,000	N/A	MAP.1301	Drop over vessel would not damage vessel and fuel.
6. Steam Dryer and Strongback	114,000	Dryer-Separator Strongback	MAP.1301	Drop over the reactor could possibly cause damage to the vessel and fuel. Further analysis is required.
7. Separator and Strongback	76,000	Dryer-Separator Strongback	MAP.1301	Drop over the reactor would not cause damage to the vessel and fuel.
8. Refueling Chute	36,000	--	Refueling Procedure	Refueling procedure will preclude load path over reactor.

a. No special lift device identified by applicant.

2.3 General Guidelines

This section addresses the extent to which the applicable handling systems comply with the general guidelines of NUREG-0612 Article 5.1.1. EG&G's conclusions and recommendations are provided in summaries for each guideline.

The NRC has established seven general guidelines which must be met in order to provide the defense-in-depth approach for the handling of heavy loads. These guidelines consist of the following criteria from Section 5.1.1 of NUREG-0612:

- Guideline 1--Safe Load Paths
- Guideline 2--Load-Handling Procedures
- Guideline 3--Crane Operator Training
- Guideline 4--Special Lifting Devices
- Guideline 5--Lifting Devices (not specially designed)
- Guideline 6--Cranes (Inspection, Testing, and Maintenance)
- Guideline 7--Crane Design.

These seven guidelines should be satisfied for all overhead handling systems and programs in order to handle heavy loads in the vicinity of the reactor vessel, near spent fuel in the spent-fuel pool, or in other areas where a load drop may damage safe shutdown systems. The succeeding paragraphs address the guidelines individually.

2.3. Safe Load Paths [Guideline 1, NUREG-0612, Article 5.1.1(1)]

"Safe load paths should be defined for the movement of heavy loads to minimize the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment. The path should follow, to the extent practical, structural floor members, beams, etc., such that if the load is dropped, the structure is more likely to withstand the impact. These load paths should be defined in procedures, shown on equipment layout drawings, and clearly marked on the floor in the area where the load is to be handled. Deviations from defined load paths should require written alternative procedures approved by the plant safety review committee."

A. Summary of Applicant's Statements

Reference [10] contains "information packets that were assembled for each elevation or section of the plant. The packets consist of general arrangement drawings that show:

(1) Location of equipment necessary for safe shutdown and continued decay heat removal with the respective emergency power division

(2) Coverage areas for the lifting devices

(3) Individual transport paths, both as elevation lifts and along the floor via dollies.

. . . Where necessary, piping composite drawings have been added to better show critical safety piping. The results of the heavy load movement study are listed in tables accompanying each packet . . . a separate table is presented for each lifting device."

In response [11], the applicant stated:

(1) Load paths will be clearly marked on the floor where the load is to be handled and appropriate procedures will be implemented. The load will be walked through by the lift supervisor only in areas where it is difficult to clearly mark the load path on the floor.

(2) No alternate load paths are presently defined in the Perry Equipment removal scheme; however, it is currently under evaluation. The plant operation review committee will designate individuals authorized to approve procedural variations, typically the maintenance or shift supervisors.

(3) Analysis and criteria used for hazard elimination, the PNPP Equipment Removal Scheme, and special Handling/Safe Load Path procedures will be maintained on file and available for review.

B. EG&G Evaluation

EG&G finds CEICo's identification of the various load paths to be very detailed and well presented. Included drawings are very informative and most of the hazard elimination categories are self-explanatory and readily acceptable.

C. EG&G Conclusions and Recommendations

Perry Nuclear Power Plant, Units 1 and 2 comply with the criteria of NUREG-0612, Section 5.1.1(1), Safe Load Paths.

2.3.2 Load-Handling Procedures [Guideline 2, NUREG-0612, Article 5.1.1(2)]

"Procedures should be developed to cover load-handling operations for heavy loads that are or could be handled over or in proximity to irradiated fuel or safe shutdown equipment. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path; and other special precautions."

A. Summary of Applicant's Statements

The applicant provided the following response in Reference [11].

"The load-handling procedures are being written and will be open for review on-site when they are complete. Attached is a draft copy of Control of Heavy Loads Procedure MAP-1301. Additional procedures on crane operating guidelines, and guidelines for rigging are provided for your information. Dollies will be used in the movement of heavy loads, but will not be pulled by cranes, i.e., cranes are not used as mechanical mules."

B. EG&G Evaluation

With the applicant preparing the necessary load-handling procedures, EG&G concludes that the criteria of Guideline 2 will be accomplished. Draft Procedure MAP-1301 specifically addresses those load movements requiring administrative procedure control as identified in the summary conclusion of the PNPP Control of Heavy Loads Study. The appplicant states (in MAP-1301) that the handling of Category A and B items will be in accordance with written-approved procedures and associated instructions or drawings. Items that are to be included in the procedures address identification of required equipment, inspections, and acceptance criteria required before movement of the load, the steps and proper sequence to be followed in handling the load, and special precautions. MAP-1301 also addresses safe load paths. The crane operating guidelines require performance of a daily prior-to-use inspection.

C. EG&G Conclusions and Recommendations

EG&G concludes that Perry Nuclear Power Plant, Units 1 and 2 are in full compliance with the criteria of NUREG-0612, Section 5.1.1(2), Load-Handling Procedures.

2.3.3 Crane Operator Training [Guideline 3, NUREG-0612, Article 5.1.1(3)]

"Crane operators should be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, 'Overhead and Gantry Cranes' [6].

A. Summary of Applicant's Statements

An overhead Crane Operator Qualification guide (MAP-0201) has been written and a draft copy provided by the applicant in Reference [9]. This guide states that operators shall be trained to ANSI B30.2-1976, ASME/ANSI N45.2.15-1981, and ANSI B30.9-1971. The Qualifications guide contains reference to NUREG-0612 for operation of the Polar, Fuel Handlings, and ESW Cranes. The following guidelines from NUREG-0612 are listed in the qualifications guide: "Operator qualification will require familiarity with the PNPP Equipment Removal Scheme and Special handling safe load path procedures. Knowledge will be checked by interviews, written exam, and practical demonstration. Completion of these requirements will be documented on a qualification card which will be reformed in the operations training file."

B. EG&G Evaluation

EG&G has determined that the program proposed by CEICo is very commendable.

C. EG&G Conclusions and Recommendations

EG&G concludes that Perry Nuclear Power Plant, Units 1 and 2 are in compliance with the criteria of NUREG-0612, Section 5.1.1(3), Crane Operator Training.

2.3.4 Special Lifting Devices [Guideline 4, NUREG-0612, Article 5.1.1(4)]

"Special lifting devices should satisfy the guidelines of ANSI N14.6-1978, 'Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials' [7]. This standard should apply to all special lifting devices which carry heavy loads in areas as defined above. For operating plants, certain inspections and load tests may be accepted in lieu of certain material requirements in the standard. In addition, the stress design factor stated in Section 3.2.1.1 of ANSI N14.6 should be based on the combined maximum static and dynamic loads that could be imparted on the handling device based on characteristics of the crane which will be used. This is in lieu of the guideline in Section 3.2.1.1 of ANSI N14.6 which bases the stress design factor on only the weight (static load) or the load and of the intervening components of the special handling device."

A. Summary of Applicant's Statements

The applicant, in response [9], submitted a draft copy of MAP-1301, a procedure for the control of Heavy Loads. Section 6.1.3.1 states "Special Lifting devices used for the movement of Heavy Loads shall meet the requirements stated in NUREG-0612, Article 5.1.1 guideline 4 and ANSI N14.6-1978."

B. EG&G Evaluation

EG&G considers the contents of draft procedures provided as being commitments toward compliance; therefore, on the basis of the applicant's statement, EG&G concludes that CEICo intends to comply with the NUREG-0612 criteria.

C. EG&G Conclusions and Recommendations

EG&G concludes that Perry Nuclear Power Plant, Units 1 and 2 are in compliance with the criteria of NUREG-0612, guideline 4, Special Lift Devices.

2.3.5 Lifting Devices (Not Specially Designed) [Guideline 5, NUREG-0612, Article 5.1.1(5)]

"Lifting devices that are not specially designed should be installed and used in accordance with the guidelines of ANSI B30.9-1971, 'Slings' [8]. However, in selecting the proper sling, the load used should be the sum of the static and maximum dynamic load. The rating identified on the sling should be in terms of the 'static load' which produces the maximum static and dynamic load. Where this restricts slings to use on only certain cranes, the slings should be clearly marked as to the cranes with which they may be used."

A. Summary of Applicant's Statements

The applicant, in response [9], submitted a draft copy of MAP-1301, a procedure for the control of Heavy Loads. Section 6.1.3.2 states "Lifting devices that are not specially designed, used for the movements of 'Heavy Loads,' shall meet the requirement stated in NUREG-0612, Article 5.1.1, guideline 5 and ANSI B30.9-1971." CEICo further stated "Each sling will be properly identified as to its lifting capacity, applicability to specific load-handling operations, and if appropriate, restriction of its use to specific cranes."

B. EG&G Evaluation

EG&G considers the contents of draft procedures provided as being commitments toward compliance; therefore, on the basis of the applicant's statement, EG&G concludes that CEICo intends to comply with the NUREG-0612 criteria.

C. EG&G Conclusions and Recommendations

EG&G concludes that Perry Nuclear Power Plant, Units 1 and 2 are in compliance with the criteria of NUREG-0612, guideline 5, Lift Devices Not Specially Designed.

2.3.6 Cranes (Inspection, Testing, and Maintenance) [Guideline 6, NUREG-0612, Article 5.1.1(6)]

"The crane should be inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' with the exception that tests and inspections should be performed prior to use where it is not practical to meet the frequencies of ANSI B30.2 for periodic inspection and test, or where frequency of crane use is less than the specified inspection and test frequency (e.g., the polar crane inside a PWR containment may only be used every 12 to 18 months during refueling operations, and is generally not accessible during power operation. ANSI B30.2, however, calls for certain inspections to be performed daily or monthly. For such cranes having limited usage, the inspections, test, and maintenance should be performed prior to their use)."

A. Summary of Applicant's Statements

"The crane inspection, testing, and maintenance program is written in detail in the Perry Maintenance Section procedures. For example, the Reactor Polar Crane monthly Preventative Maintenance, Reactor Polar Crane Quarterly and semiannual Preventive Maintenance, and Reactor Polar Crane Yearly Preventative Maintenance procedure (see attached) are written in detail and Reference ANSI B30.2-1976. CEI will meet the specifications of ANSI B30.2-1976" [9].

B. EG&G Evaluation

The draft copy of the PNPP Preventative Maintenance Instructions appear to be very complete and well presented. EG&G considers the content of draft procedures provided as being commitment toward compliance; therefore, on the basis of the applicant's statement, EG&G concludes that CEICo intends to comply with the NUREG-0612 criteria.

C. EG&G Conclusions and Recommendations

EG&G concludes that Perry Nuclear Power Plant, Units 1 and 2 are in compliance with the criteria of NUREG-0612, guideline 6, Cranes (Inspection, Testing, and Maintenance).

2.3.7 Crane Design [Guideline 7, NUREG-0612, Article 5.1.1(7)]

"The crane should be designed to meet the applicable criteria and guidelines of Chapter 2-1 of ANSI B30.2-1976, 'Overhead and Gantry Cranes,' and of CMAA-70, 'Specifications for Electric Overhead Traveling Cranes' [9]. An alternative to a specification in ANSI B30.2 or CMAA-70 may be accepted in lieu of specific compliance if the intent of the specification is satisfied."

A. Summary of Applicant's Statements

"All Cranes identified in this report that handle loads in safety-related buildings are designed to CMAA Specification 70 and ANSI B30.2-1976 [10]." The applicant provided additional information in Reference [9] that includes crane nufacturer, type, serial number, load capacity, and purchase data.

B. EG&G Evaluation

On the basis of the applicant's statement and the additional information provided, EG&G feels that the criteria of guideline 7 will be satisfied.

C. EG&G Conclusions and Recommendations

EG&G concludes that Perry Nuclear Power Plant, Units 1 and 2 are in compliance with the criteria of NUREG-0612, guideline 7, Crane Design.

2.4 Interim Protection Measures

The NRC staff has established (NUREG-0612, Article 5.3) that six measures should be initiated to provide reasonable assurance that handling of heavy loads will be performed in a safe manner until final implementation of the general guidelines of NUREG-0612, Article 5.1, is complete. Four of these six interim measures consist of general guideline 1, Safe Load paths; guideline 2, Load-Handling Procedures; guideline 3, Crane Operator Training; and guideline 6, Cranes (Inspection, Testing, and Maintenance). The two remaining interim measures cover the following criteria:

- Heavy load technical specifications
- Special review for heavy loads handled over the core.

Applicant implementation and evaluation of these interim protection measures is contained in the succeeding paragraphs of this section.

EG&G recommends that because CEICo Perry Nuclear Power Plant, Units 1 and 2 are not yet operational, and will not be operational for quite some time, it is more appropriate that the time be spent completing those commitments toward compliance with the guidelines in NUREG-0612, Article 5.1 rather than addressing interim measures. Proper compliance with these guidelines negates the necessity of interim protection measures.

CEICo stated that their present position is to be in full compliance with the guidelines of NUREG-0612 prior to fuel load [9].

3. CONCLUDING SUMMARY

3.1 Applicable Load-Handling Systems

The list of cranes and hoists supplied by the applicant as being subject to the provisions of NUREG-0612 is complete (see Section 2.2.1). The Applicant also fulfilled the requirements of NUREG-0612 concerning exclusion of various Overhead Hanc`ing Systems.

3.2 Guideline Recommendations

Compliance with the seven NRC guidelines for heavy load-nandling (Section 2.3) are satisfied at Perry Nuclear Power Plant, Units 1 and 2. This conclusion is represented in tabular form as Table 3.1.

TABLE 3.1. COMPLIANCE MATRIX PERRY NUCLEAR POWER PLANTS, UNITS 1 AND 2

Equipment Designation	Heavy	Capacity (lb)	Guideline 1 Safe Load Paths	Guideline 2 Procedures	Guideline 3 Crane Operator Training	Guideline 4 Special Lift Devices	Guideline 5 Slings	Guideline 6 Crane-Test and Inspection	Guideline 7 Crane Design
Emergency Service Water Pump House Crane	C	30,000	C	C	C	C	C	C	C
Reactor Building Crane	C	150,000	C	C	C	C	C	C	C
Fuel-Handling Building Crane	C	250,000	C	C	C	C	C	C	C

C=Applicant action complies with NUREG-0612 guideline.

4. REFERENCES

1. NUREG-0612, Control of Heavy Loads at Nuclear Power Plants, NRC.

2. V. Stello, Jr. (NRC), Letter to all applicants. Subject: Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 17 May 1978.

3. USNRC, Letter to CEICo. Subject: NRC Request for Additional Information on Control of Heavy Loads Near Spent Fuel, NRC, 22 December 1980.

4. Letter to NRC, Subject: Control of Heavy Load Study from D. R. Davidson, Vice President, The Cleveland Electric Illuminating Company to D. G. Eisenhut, Director, USNRC, 19 June 1981.

5. ANSI B30.2-1976, "Overhead and Gantry Cranes".

6. ANSI N14.6-1978, "Standard for Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or more for Nuclear Materials".

7. ANSI B30.9-1971, "Slings".

8. CMAA-70, "Specifications for Electric Overhead Traveling Cranes".

9. Letter to NRC, Subject: Response to Draft Technical Evaluation Report. Control of Heavy Loads, from D. R. Davidson, The Cleveland Electric Illuminating Company to A. Schwencer, USNRC, 15 September 1982.

10. The Cleveland Electric Illuminating Company GAI Report No. 2329, Rev. 2 (7 January 1982), Control of Heavy Load Study.

11. Letter to NRC, Subject: Revision of Response to Technical Evaluation Report--Control of Heavy Loads, D. R. Davidson, Cleveland Electric Illuminating Company to A. Schwencer, USNRC, 8 November 1982.

APPENDIX L

FEDERAL EMERGENCY MANAGEMENT AGENCY
INTERIM REPORT ON OFFSITE RADIOLOGICAL EMERGENCY PLANNING
FOR THE PERRY NUCLEAR POWER STATION

Federal Emergency Management Agency

Washington, D.C. 20472

MAR 1 1984

MEMORANDUM FOR: Edward L. Jordan
Director, Division of Emergency Preparedness and Engineering Response
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission

FROM: Richard W. Krimm
Assistant Associate Director
Office of Natural and Technological Hazards Programs

SUBJECT: Interim Report on Offsite Radiological Emergency Planning for the Perry Nuclear Power Station

Attached is an interim report on offsite radiological emergency planning for the Perry Nuclear Power Station. The report, dated January 10, 1984, was prepared by Region V of the Federal Emergency Management Agency (FEMA). Since an exercise and public meeting have not yet been accomplished in accordance with 44 CFR Part 350, the report primarily addresses the review process of the State and County plans. Included in this report also are responses to requests from civic groups, interrogatories with comments, newspaper releases, and a brief geographic "sketch" of the emergency planning zone.

A number of planning deficiencies were identified by the Region V Assistance Committee during their informal review of the first draft of the county plans on April 28, 1983, and during their formal review of the Ohio State Radiological Emergency Preparedness Plan on June 3, 1980.

The Ohio Disaster Services Agency has provided a schedule of corrective actions indicating the manner in which the county plans are to be revised to correct the deficiencies. Based on the Region V review of the Ohio State and Ashtabula, Geauga, and Lake Counties offsite radiological emergency preparedness plans, there is reasonable assurance that the plans are adequate and capable of being implemented in the event of an accident at the site. An exercise to test these plans is scheduled for November 28, 1984. A finding on preparedness will be made following this exercise.

If you have any questions relative to this report, please feel free to contact Mr. Marshall Sanders, Acting Chief, Technological Hazards Division, at 287-0179.

Attachment
As Stated

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Federal Emergency Management Agency

Region V 300 South Wacker, 24th Floor, Chicago, IL 60606 (312) 353-1500

February 6, 1984

Memorandum For: Assistant Associate Director, Office of Natural and Technological Hazards

Attn: Robert Turner - SL-NT-NH

From: Edward J. Roche, Sr. Regional Director Edward J. Roche Sr.

Subject: Perry Interim Finding Conclusion

The Region V Assistance Committee (RAC) completed their informal review of the first (1st) draft of the Ashtabula, Geauga, and Lake County, Ohio, Radiological Emergency Preparedness plan on April 28, 1983. The State of Ohio Radiological Emergency Preparedness plan was formally reviewed by the RAC on June 3, 1980. These reviews, utilizing NUREG 0654/FEMA REP-1, Revision 1, planning criteria, identified a number of planning deficiencies.

The State of Ohio REP plan has been exercised five (5) times during both full and partial exercises for the Beaver Valley, Davis Besse, and Zimmer nuclear power plants. The first joint full participation exercise between the utility, the State and Ashtabula, Geauga, and Lake Counties is scheduled for November 28, 1984.

The Ohio Disaster Services Agency (ODSA) has received the RAC review for the off-site specific plan for Perry and has provided a schedule of corrective actions indicating the manner the County plans are to be revised to correct the deficiencies.

Given the above, FEMA Region V concludes that the remaining deficiencies, considered as a whole, are such that, in spite of them, there is reasonable assurance that appropriate protective measures can be taken in the event of a radiological emergency at the Perry Nuclear Power Plant. This conclusion is based solely on the basis of a plan review. Further evaluation of State and local governments ability to implement these plans will be made as a result of the November 28, 1984, full participation exercise.

NRC FORM 335 (6-83) U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)	NUREG-0887 Supplement No. 5
2. Leave blank	
3. TITLE AND SUBTITLE	Safety Evaluation Report related to the operation of the Perry Nuclear Power Plant, Units 1 and 2
4. RECIPIENT'S ACCESSION NUMBER	
5. DATE REPORT COMPLETED	MONTH: FEBRUARY YEAR: 1985
6. AUTHOR(S)	
7. DATE REPORT ISSUED	MONTH: FEBRUARY YEAR: 1985
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12a. TYPE OF REPORT	Technical
12b. PERIOD COVERED (Inclusive dates)	March 1984 - February 1985
13. SUPPLEMENTARY NOTES	Docket Nos. 50-440 and 50-441

14. ABSTRACT (200 words or less)

Supplement No. 5 to the Safety Evaluation Report (NUREG-0887) pertains to the application filed by the Cleveland Electric Illuminating Company on behalf of itself and as agent for the Duquesne Light Company, the Ohio Edison Company, the Pennsylvania Power Company, and the Toledo Edison Company (the Central Area Power Coordination Group or CAPCO), as applicants and owners, for a license to operate the Perry Nuclear Power Plant, Units 1 and 2 (Docket Nos. 50-440 and 50-441). The report has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in Lake County, Ohio, approximately 35 miles northeast of Cleveland, Ohio. This supplement reports the status of certain issues that had not been resolved at the time of publication of the Safety Evaluation Report and Supplement Nos. 1 through 4.

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