NUREG-0887
Supplement No. 2

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

January 1983

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ABSTRACT

Supplement No. 2 to the Safety Evaluation Report 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 (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. 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 No. 1 to that report.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ADS	automatic depressurization system
AISC	American Institute of Steel Construction
ALARA	as low as is reasonably achievable
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient(s) without scram
BOP	balance of plant
BTP	Branch Technical Position
BWR	boiling-water reactor
CAPCO	Central Area Power Coordination Group
CEI	Cleveland Electric Illuminating Company
CFR	Code of Federal Regulations
CP	construction permit
CRD	control rod drive
DCRDR	detailed control room design review
ECCS	emergency core cooling system
EHC	electrohydraulic control
ESF	engineered safety feature(s)
FSAR	Final Safety Analysis Report
GDC	General Design Criterion(a)
GE	General Electric
HCU	hydraulic control unit
HED	human engineering discrepancy
HPCS	high-pressure core spray
IE	Office of Inspection and Enforcement
IFTS	inclined fuel transfer system
INPO	Institute of Nuclear Power Operations
L_a	maximum allowable leakage rate at pressure P_a
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LRG-II	Licensing Review Group-II
MCPR	minimum critical power ratio
MSIV	main steam isolation valve
MWd/MTU	megawatt days per metric ton of uranium
NIOSH	National Institute of Occupational Safety and Health
NPTS	Nuclear Project Training Section
NSSS	nuclear steam supply system
OL	operating license
OSHA	Occupational Safety and Health Act
P_a	calculated peak containment pressure
PGCC	power generation control complex
PPDTU	Perry Plant Department Training Unit
psig	pounds per square inch gage
QA	quality assurance

RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RV	relief valve
SDV	scram discharge volume
SER	Safety Evaluation Report
SSER	Supplemental Safety Evaluation Report
SRV	safety relief valve
SV	safety valve
TMI	Three Mile Island

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Nuclear Regulatory Commission's 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, Units 1 and 2, was issued in May 1982. Supplement No. 1 to the Safety Evaluation Report (SER) was issued in August 1982. The purpose of this supplement is to further update the SER by providing the results of the staff's review of information submitted by the applicant by letter or in meetings to address some of the issues in Sections 1.9, 1.10, and 1.11 of the SER that remain unresolved. The information provided in the applicant's letters must be acceptably documented in Amendments to the Final Safety Analysis Report (FSAR) before licensing.

Each section or appendix of this Supplemental Safety Evaluation Report (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 NRC and the applicant that updates the list in the SER and SSER No. 1. Appendix B is a list of references cited in this report.* Appendix C addresses the staff's resolution of Unresolved Safety Issue, Task A-11. Appendix E is a list of principal staff contributors to this supplement. Appendix G is a list of additional errata to the SER. No changes were made to SER Appendices D or F in this supplement.

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

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

*Availability of all material cited is described 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 in May 1982. Three of those issues were reported as satisfactorily resolved and Issue (20) was added in SSER No. 1. This supplement discusses those issues that have been resolved since SSER No. 1 was issued in August 1982, as well as the status of those issues that continue to remain unresolved. Outstanding Issue (3) has been changed to Confirmatory Issue (53). An additional outstanding issue is being added in this supplement as Issue (21), namely, "Reanalysis of transients and accidents: development of emergency operating procedures per TMI Action Plan Item I.C.1 (13.5.2.1)." The 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 supplement to the SER.

Issue	Status	Section
(1) Turbine missile protection	Under review; awaiting additional information	--
(2) Seismic system and subsystem analysis	Resolved in SSER No. 1	--
(3) Reactor internal vibration prototype (BWR/6-238 in.) test program	Changed to Confirmatory Issue (53)	3.9.2.3
(4) Environmental/seismic and dynamic qualification of Category I mechanical and electrical equipment	Under review	--
(5) Inservice testing of pumps and valves	Awaiting information	--
(6) Transient and accident analysis for ECCS, overpressure, and operating MCPR	Resolved in SSER No. 1	--
(7) Control room design	Interim DCRDR audit performed	18
(8) Mark III containment system issues (Humphrey issues)	Awaiting information	--
(9) Pool dynamic loads	Awaiting information	--
(10) Containment purge	Under review	--
(11) Periodic testing of ADS actuation systems during plant operation	Resolved	7.3.2.6
(12) Manual initiation/termination of ESF systems	Review progress update - awaiting information	7.3.2.7

Issue	Status	Section
(13) IE Bulletin 79-27	Awaiting information	--
(14) Control system failures	Awaiting information	--
(15) Fire protection - safe shutdown	Resolved	9.5.1.4.2
(16) Fire protection - PGCC system (CO_2 vs Halon fire suppressant in control room)	Resolved	9.5.1.6.2
(17) HPCS engine skid piping	Resolved in SSER No. 1	--
(18) Interim shifting staffing for two-unit operation	Deferred; applies to Unit 2 only	--
(19) Emergency plans	Awaiting information	--
(20) Standby liquid control system final design	Added in SSER No. 1 - update on revised system design status; awaiting information	9.3.4
(21) Reanalysis of transients and accidents: development of emergency operating procedures per TMI Action Plan Item I.C.1	Awaiting information	--

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. Five of those issues were resolved and Issue (50) was added in SSER No. 1 (Issue (50) was cited as License Condition (8) in the SER). This supplement discusses those issues that have been resolved since SSER No. 1 was issued and adds three new issues - (51), (52), and (53). The status of each confirmatory issue is addressed below. If the issue is discussed in this supplement, the section where it is discussed is noted. Resolution of the remaining confirmatory issues will be addressed in a future supplement to the SER.

Issue	Status	Section
(1) Piping final stress analysis	Awaiting information	--
(2) Containment buckling analysis	Resolved in SSER No. 1	--
(3) Containment ultimate capacity analysis	Resolved in SSER No. 1	--
(4) Emergency service water tunnel structure analysis	Resolved in SSER No. 1	--

Issue	Status	Section
(5) Vibration monitoring program for BOP systems	Resolved in SSER No. 1	--
(6) Mark III hydrodynamic loads	Deleted - same as Outstanding Issue (9) in Section 1.9	--
(7) Testing relief/safety valves per TMI Action Plan Item II.D.1	Under review	--
(8) IE Bulletin 79-02	Under review	--
(9) Dual function pipe whip/support restraints	Resolved	3.9.3.3
(10) Hydrodynamic load effect on CRD/HCU	Under review	--
(11) Fuel mechanical fracturing	Under review	--
(12) Fuel assembly damage from external sources	Under review	--
(13) Fuel rod bowing	Under review	--
(14) Overheating of gladolinia fuel pellets	Under review	--
(15) Preservice inspection program	Awaiting information	--
(16) Material surveillance capsules - RV beltline	Resolved	5.3.1(2)
(17) Fracture toughness RCPB materials	Resolved - site confirmatory audit required before fuel load	--
(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	Under review	--
(21) Suppression pool temperature limits	Awaiting information	--
(22) Secondary containment penetration leakage	Resolved	6.2.1.9
(23) Containment isolation dependability per TMI Action Plan Item II.E.4.2	Resolved	6.2.7(3)
(24) Type C test of all ECCS injection valves	Resolved	6.2.6

Issue	Status	Section
(25) ADS logic modification per TMI Action Plan Item II.K.3.18	Awaiting information - also added as License Condition (17) in this supplement	--
(26) ATWS recirculating pump trip	Awaiting information	--
(27) Modified SDV level monitoring system	Resolved	7.2.2.4
(28) HPCS initiation circuitry final design	Resolved - site confirmatory audit required before fuel load	7.3.2.2
(29) Remote shutdown panel nonsafety-grade readouts	Resolved	7.4.2.2
(30) RCIC testing procedures	Resolved	7.4.2.4
(31) Calibration for RV/SV pressure switches	Resolved - site confirmatory audit required before fuel load	SER 7.5.2.1
(32) Accident monitoring per TMI Action Plan Items II.F.1.4, II.F.1.5, and II.F.1.6	Resolved	7.5.2.5
(33) Failures in vessel level sensing lines common to control and reactor protection systems	Resolved	7.7.2.3
(34) Final valve design setpoint and analysis	Resolved	8.2.4
(35) Physical separation of redundant electrical systems	Resolved - site confirmatory audit required before fuel load	--
(36) Documentation or test of 3-hour-fire resistance of gypsum board walls	Test required to resolve - awaiting information	--
(37) Light and communication fire protection features	Under review	--
(38) Revision of fire protection stand-pipe and hose locations	Under review	--
(39) Portable fire extinguisher locations	Under review	--
(40) Watertight curbs in switchgear/ diesel generator rooms	Under review	--
(41) Design for noble gas effluent monitors per TMI Action Plan Item II.F.1.1	Awaiting information	--

Issue	Status	Section
(42) Design for sampling and analysis of plant effluents per TMI Action Plan Item II.F.1.2	Awaiting information	--
(43) Leakage surveillance preventive maintenance program per TMI Action Plan Item III.D.1.1	Changed to License Condition (16) in SSER No. 1	--
(44) Radiation/shielding design of IFTS tube	Resolved	12.3.2
(45) Location of plant area radiation monitoring per TMI Action Plan Item II.F.1.3	Resolved	12.3.4.1
(46) Training program per TMI Action Plan Item II.B.4	Resolved	13.2.1.9
(47) Nuclear section training program	Resolved	13.2.2.1
(48) Shift supervisor training per TMI Action Plan Item I.C.3	Resolved	13.5.1.8
(49) Verify implementation of equipment control measures in radiation areas per TMI Action Plan Item I.C.6	Resolved	13.5.1.11
(50) No load, light load, and test loading of the diesel generators	Resolved	9.6.3.1(4)
(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	Awaiting information	--
(53) Reactor internals vibration prototype (BWR/6-238 in.) test program	Staff needs to complete review of GE program adopted by CEI to resolve	3.9.2.3

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 an asterisk) that will be cited in the operating license issued, to ensure that NRC requirements are met during plant operation. License Condition (8) was deleted from this section and added to the list of confirmatory issues as Confirmatory Issue (50) in Section 1.10 of SSER No. 1. This supplement adds License Condition (17), previously listed in Section 1.10 as Confirmatory Issue (25), and License Condition (18), "Control of heavy loads." In addition, the staff has completed its generic evaluation of License Condition (2), which is addressed in Section 4.2.3.1 of this supplement.

The updated and current list of license conditions, with references to appropriate SER sections, follows:

(1) Implementation of protective measures when the Lake Erie shoreline recedes to 250 ft from the emergency service water pumphouse.*

(2) Periodic measurement of channel box deflections must be resolved before startup of the second cycle of operation - see Section 4.2.3.1 of this supplement for results of the staff's generic evaluation of this license condition.

(3) Operation beyond Cycle 1 not permitted until thermal-hydraulic stability analyses are provided for approval before Cycle 2 operation.*

(4) A final report analyzing inadequate core cooling implementation requirements per TMI Action Plan Item II.F.2 should be submitted for staff approval.*

(5) Hydrogen control for degraded core accidents per TMI Action Plan Item II.B.8 subject to completion of staff generic evaluation.*

(6) IE Bulletin 80-06, engineered safety feature reset control.

(7) Postaccident sampling system per TMI Action Plan Item II.B.3.

(8) No load, light load, and test loading of the diesel generators - changed to Confirmatory Issue (50) in SSER No. 1 - is no longer at issue and will be deleted in future supplements to the SER (see Sections 1.10 and 9.6.3.1(4) of this supplement).

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

(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 a nominal 100% power, whichever occurs later.*

(11) Test and maintenance procedures associated with engineered safety features per TMI Action Plan Item II.K.1.15.

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

(13) Complete implementation and maintenance of staff-approved physical security, guard training and qualification, and safeguards contingency plans.*

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

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

(16) Leakage surveillance and preventive maintenance program per TMI Action Plan Item III.D.1.1.

(17) ADS logic modification per TMI Action Plan Item II.K.3.18 - installation of approved modification is required before plant startup after the first refueling outage.*

(18) Before startup following the second refueling outage, the applicant shall have made commitments acceptable to the NRC regarding the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II - 9-month responses to the NRC generic letter dated December 22, 1980).

Finally, an additional Technical Specification item is being included to require the applicant to perform periodic surveillance tests and calibration of the low-pressure air alarm systems. This is discussed in Sections 6.3.1.3 and 16 of this supplement.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.9 Mechanical Systems and Components

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

3.9.2.3 Preoperational Flow-Induced Vibration Testing of Reactor Internals

In Section 3.9.2.3 of the SER, the staff identified, as Outstanding Issue (3), the reactor internals vibration test program for a prototype reactor (BWR/6-238 in.). The staff required that the applicant submit a prototype test program fully consistent with Regulatory Guide 1.20. The staff also required that the resolution of past boiling-water-reactor (BWR) problems be addressed by the applicant's test program.

It has been determined that the applicant adequately addressed the resolution of past BWR problems such as the degradation of feedwater spargers, fuel box channel wear, and jet pump holddown beams (discussed in SER Sections 3.9.3.1, 4.2.3.1(5), and 3.9.5, respectively). The applicant's response regarding analytical predictions of internals vibration levels was addressed in a letter dated September 9, 1982 (D. R. Davidson to A. Schwencer), which included a draft copy of General Electric (GE) Topical Report NEDE-22203-P, entitled "Reactor Internals Vibration Predictions." The report describes the GE reactor internals vibration test program to be adopted by CEI and provides peak amplitude predictions obtained from engineering models and prototype reactor test data.

The staff has not yet completed its review of the GE report. Therefore, although the staff concludes that the applicant has satisfactorily responded to Regulatory Guide 1.20 in providing a complete prototype test program, this issue cannot be fully resolved until the staff completes its review of the GE report data applicable to Perry. On the basis of additional information provided, Outstanding Issue (3), listed in Section 1.9 of the SER, is now considered to be confirmatory and is being added to Section 1.10 of the SER as Confirmatory Issue (53) in this supplement. Its resolution will be addressed in a future supplement to the SER.

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

3.9.3.3 Component Supports

In Section 3.9.3.3 of the SER, the staff identified the design of pipe whip restraints, which also function as a pipe support, as a confirmatory issue. The applicant responded to this issue in a letter dated April 16, 1982 (D. R. Davidson to A. Schwencer), which identifies those restraints that have a dual pipe whip restraint/support function, and which describes the boundary used for determining its classification under Subsection NF, Section III of the American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code" (ASME Code) and the American Institute of Steel Construction (AISC) building specification.

The staff is currently developing a generic position on the classifications of auxiliary steel used as a pipe support (Subsection NF, Section III of the ASME Code) or building steel (AISC specification), as well a the plant inservice inspection requirements. The staff finds that the classification for the dual pipe whip restraint/support for the Perry design is consistent with currently accepted industry practices and with the generic position being developed by the staff. Therefore, Confirmatory Issue (9), listed in Section 1.10 of the SER, is considered to be resolved.

4 REACTOR

4.2 Fuel System Design

4.2.3 Design Evaluation

4.2.3.1 Fuel System Damage Evaluation

(5) Dimensional Changes

Channel Box Deflection (LRG-II Generic Issue 3-CPB)

The BWR fuel channels provide structural stiffness for the fuel assemblies and distribute the coolant flow between the assemblies and channel bypass regions. The channels are subject to time-dependent, permanent dimensional changes (i.e., deflections) that result from irradiation, creep, and stress-relaxation effects. The resultant bulge (caused by long-term thermal creep) or bow (caused by differential irradiation-induced axial growth) reduces the size of the gap available for control rod insertion. Channel box deflection is thus a potential life-limiting phenomenon.

In Section 4.2.3.1(5) of the SER, the staff indicated that the issue of channel box deflection was resolved for Zimmer and other BWRs through the staff's acceptance of the channel box surveillance plan proposed by the plants' licensees. In a letter dated November 3, 1981 (D. G. Eisenhut to D. R. Davidson), the staff notified CEI of this and requested CEI to commit to the surveillance plan approved for those plants. In a letter dated May 17, 1982 (D. L. Holtzscher to H. J. Faulkner), the Licensing Review Group-II (LRG-II), a group in which CEI is a participant, submitted a position paper (addressing LRG-II generic issue 3-CPB) on channel box deflection surveillance measures. This position and the surveillance measures proposed (adopted by CEI for Perry in a letter dated September 16, 1982) use several of the same surveillance measures approved by the staff for Zimmer and other BWR plants, namely:

(a) Records will be kept of channel locations and exposure for each operation cycle.

(b) Channels shall not reside in the outer row of the core for more than two operating cycles (because flux gradients are largest near the core periphery, and, therefore, differential irradiation-induced growth and bowing will be greatest at those locations).

(c) At the beginning of each fuel cycle, the combined outer-row residence time for any two channels in any control rod cell shall not exceed four peripheral cycles.

In addition, channels that reside in the periphery (outer row) for more than one cycle shall be situated each successive peripheral cycle in a location that rotates the channel so that a different side faces the core edge. The

staff believes that this should help to lessen channel bowing and that measures given in Items (a), (b), and (c) above would also help to reduce the magnitude of channel deflection.

The LRG-II position paper endorsed by CEI for Perry further provides a detailed description of the test program for the control rod drive settling function, which will be performed for any core cells that exceed the above measures, or which contain channels with exposures greater than 30,000 MWd/MTU (associated fuel bundle exposures).

The staff finds that CEI's commitment to these LRG-II measures and test program would preclude excessive channel bowing in Perry, and concludes that License Condition (2) pertaining to the need to periodically measure channel box deflections before startup of the second cycle of operation has been satisfactorily addressed. These measures and test program will be appropriately cited in the operating license for Perry. The staff will continue to review this phenomenon generically with General Electric. Should the staff's generic findings warrant further measures for application to Perry, CEI will be advised accordingly.

5 REACTOR COOLANT SYSTEM

5.3 Reactor Vessel Materials, Fabrication, and Integrity

5.3.1 Reactor Vessel Materials

(1) Compliance With Appendix G, 10 CFR 50

In Section 5.3.1(1) of the SER, the staff reported that it had reviewed the applicant's FSAR to determine the degree of compliance with the fracture toughness requirements of 10 CFR 50, Appendix G, and stated that the applicant was in compliance with Appendix G except for Paragraphs III.B.4 and IV.A.2.c. It was concluded that the applicant's proposed exemptions for performing tests in accordance with the written procedures of Paragraph III.B.4 were justifiably supported, and that the applicant's proposed alternative to Paragraph IV.A.2.c to the criticality hydrostatic temperature limit was also acceptable. Although not reported in the SER, the staff recognized that operating limits in accordance with Appendix G, Section III, of the ASME Code would not provide adequate safety margins for the Perry closure-to-flange and reactor vessel shell-to-flange discontinuities in accordance with Paragraphs IV.A.2.a and IV.A.2.b of Appendix G of 10 CFR 50.

Paragraphs IV.A.2.a and IV.A.2.b of Appendix G, 10 CFR 50, require, in part, that flange and shell regions near geometric discontinuities shall provide margins of safety in accordance with Appendix G, Section III, of the ASME Code. The applicant's fracture mechanics analysis indicates that operating limits in excess of those required by Appendix G, Section III, of the ASME Code would be required for flaw depths greater than 0.24 in. on the outside surface of the flange-to-shell joint. To satisfy this fracture mechanics evaluation the applicant has proposed either to:

(a) perform an augmented inservice examination of the flange-to-shell and head discontinuities that will detect flaws less than 0.24 in. on the outside surface, or

(b) revise the pressure-temperature limits to provide margins of safety for the flange-to-shell and head discontinuities that are equivalent to those required by Appendix G, Section III, of the ASME Code.

The staff considers that an augmented inservice examination and/or revised pressure-temperature limits can provide adequate margins of safety for the flange-to-shell and head discontinuities.

The staff will assess the applicant's proposed inservice examination requirements for the flange-to-shell and head discontinuities and the revised pressure-temperature limits during its review of the applicant's inservice examination plan and NRC's Technical Specification to ensure that safety margins, equivalent to those of Appendix G, Section III, of the ASME Code, are met.

(2) Compliance With Appendix H, 10 CFR 50

In Section 5.3.1(2) of the SER, the staff reported on its review of the applicant's compliance with 10 CFR 50, Appendix H, concluding the applicant's noncompliance with Paragraph II.B of Appendix H to be a confirmatory issue. Specifically, Paragraph II.B requires that the applicant's surveillance program comply with American Society for Testing and Materials (ASTM) E-185-73. ASTM E-185-73 requires that the materials in the surveillance capsules be removed from reactor vessel beltline base metals and weld metal samples that will be limiting for operation of the reactor vessel during its lifetime. At the time the SER was issued, the applicant's FSAR amendments through Amendment 6 had not reported the weld metal in the surveillance program.

In a letter dated September 22, 1982 (D. R. Davidson to A. Schwencer), the applicant identified the weld material in the Perry plant surveillance program. The staff's review of the surveillance program weld material indicates that the material is representative of the limiting reactor vessel beltline weld metal and satisfies the surveillance program requirements of Paragraph II.B of Appendix H. Thus, Confirmatory Issue (16), listed in Section 1.10 of the SER, is considered to be resolved.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.9 Secondary Containment

For Perry, the secondary containment structure consists of the shield building, which is a cylindrical reinforced concrete structure that completely surrounds the containment. The secondary containment is used to control and treat radioactive leakage from the primary containment in the event of a loss-of-coolant accident (LOCA). Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary and secondary containment boundaries creating potential leakage paths, in the event of a LOCA, through which radioactivity in the primary containment could bypass the leakage collection and filtration systems associated with the secondary containment.

In Section 6.2.1.9 of the SER, the staff found that the applicant had not sufficiently considered leakage from lines that penetrate the primary and secondary containment, believed to be potential bypass leakage paths. The staff considered this to be a confirmatory issue and required the applicant to justify their exclusion from leakage consideration.

In a letter dated June 7, 1982 (D. R. Davidson to A. Schwencer), the applicant addressed this issue and provided the criteria the applicant used to determine and assess potential bypass leakage paths. These criteria showed that the penetration lines in question were excluded because they contain physical barriers or design provisions (e.g., the lines contain water seals, they involve closed Category I piping systems, and/or leakage controls are provided in the design) that will effectively eliminate leakage. Where relied on to eliminate leakage, these provisions are designed to (1) meet the single-failure criteria, (2) be missile protected, and (3) have a temperature and pressure rating in excess of that for containment.

The staff finds that the additional information furnished by the applicant sufficiently justifies exclusion of these lines from bypass leakage consideration in that the physical barriers and design provisions identified will essentially eliminate any leakage and are consistent with the acceptance criteria of Branch Technical Position CSB 6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants." Therefore, Confirmatory Issue (22), listed in Section 1.10 of the SER, is considered to be resolved.

6.2.3 Containment Isolation System

This issue was inadvertently omitted from the list of confirmatory issues contained in Section 1.10 of the SER and is accordingly addressed below.

The containment isolation system includes the containment isolation valves, associated piping, and penetrations necessary to isolate the primary containment in the event of a LOCA. In Section 6.2.3 of the SER, the staff reported that the containment isolation provisions in the Perry design, for lines penetrating containment, conformed with the requirements of General Design Criteria (GDC) 55, 56, and 57 (10 CFR 50, Appendix A), as appropriate. However, the applicant had not adequately correlated all of the proposed deviations from the explicit requirements of GDC 55 and 56. (As indicated in these GDCs, there are containment penetrations whose isolation provisions do not have to satisfy the explicit requirements specified therein, but can be accepted on some other defined basis, e.g., the use of the alternative criteria in Section 6.2.4 of the Standard Review Plan (NUREG-0800)). The need for a clear correlation of the criteria used was considered to be a confirmatory issue by the staff.

In a letter dated June 7, 1982 (D. R. Davidson to A. Schwencer), the applicant furnished additional information that can distinguish isolation provisions in the plant that explicitly meet GDC 55 and 56, as well as those for which the alternative criteria in Section 6.2.4 of NUREG-0800 are appplicable. The applicant's treatment of these criteria are summarized as follows:

(1) Lines that must remain in service following an accident and lines that must remain in service during normal operation for safety reasons are provided with at least one isolation valve. A second isolation boundary is formed by a closed system outside the containment.

(2) Where a closed system outside the containment forms the second isolation boundary, each of the systems and all components that form its boundary are designed to Quality Group B and seismic Category I standards. Valves that isolate the branch lines of these closed systems outside containment are normally closed and under strict administrative control.

(3) On certain engineered safety features or related system, remote manual valves are used instead of automatic valves, since these lines must remain in service following an accident. Where remote manual valves are used, leakage-detection capabilities are provided.

(4) On some penetrations, the containment isolation provisions consist of two valves in series, both of which are outside the containment. For these penetrations, locating one of the valves inside containment would subject it to more severe environmental conditions (including suppression pool dynamic loads) than if it were outside containment; moreover, the inside valve would then not be easily accessible for inservice inspection.

On the basis of the additional information provided by the applicant and summarized above, the staff finds that the applicant satisfactorily correlates criteria used for the Perry containment isolation system design and concludes that this confirmatory issue is resolved.

6.2.6 Containment Leakage Testing

The staff reviewed the applicant's containment leak-testing program for compliance with the containment leak-testing requirements specified in Appendix J to 10 CFR 50. Such compliance provides adequate assurance that the leak-tight

integrity of the containment can be verified throughout the service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakage within the specified limits. Maintaining containment within such limits provides reasonable assurance that in the event of any radioactivity release within the containment, the loss of the containment atmosphere through potential leak paths will not be in excess of the limits specified for the site.

In its review of the applicant's leak-test program (as reported in Section 6.2.6 of the SER), the staff found that the applicant's test program ensures that containment penetrations and system isolation valve arrangements are designed to satisfy the containment integrated leak rate and the local leak-testing requirements of 10 CFR 50, Appendix J, in that the program will include an ASME Code Type C test of all emergency core cooling system injection valves with air, unless it can be demonstrated that a water seal exists that meets the single-failure criteria. However, the applicant was asked, as a confirmatory issue, to incorporate the following additional provisions in the containment leak-test program:

(1) All isolation valves listed in Table 6.2-40 of the test program should be Type C tested.

(2) The feedwater lines (test items 9 and 10) should be vented and drained for a Type C test, tested in air, and the leakage included in 0.60 L_a.

(3) High-pressure core spray (HPCS) pump discharge to the reactor vessel (test item 32) and the low-pressure core spray (LPCS) pump discharge to the reactor vessel (test item 35) should be tested with air and the leakage included in 0.60 L_a (60% of maximum allowable leakage rate at pressure P_a, which is the calculated peak containment pressure).

In a letter dated June 8, 1982 (D. R. Davidson to A. Schwencer), the applicant responded to this confirmatory issue as follows:

(1) All containment isolation valves listed in Table 6.2-40 of the test program will be Type C tested except for the instrument line isolation valves that penetrate the containment and that conform to Regulatory Guide 1.11. Isolation valves, pressurized by a water seal system, will be consistent with the Type C test acceptance criteria in Appendix J. Examples of such lines are discussed in (2) and (3) below, and are lines that terminate below the water level of the suppression pool. Sufficient pool inventory is available to maintain a 30-day pressure at 1.10 P_a.

    The piping up to each isolation valve is seismic Category I, Safety Class 2 piping - missile and pipe whip are not concerns for this piping.

(2) The feedwater lines will be Type C tested with water and the leakage will not be included in the 0.60 L_a. This is consistent with the Appendix J acceptance criteria because a dedicated feedwater leakage control system is provided for these lines.

(3) HPCS, LPCS, and low-pressure coolant injection pump discharge lines to the reactor vessel will be Type C tested with air, and the largest leakage

included in 0.60 L_a. Consistent with Appendix J acceptance criteria hydrostatic testing may be performed if a liquid inventory to maintain a water seal is demonstrated, assuming single failure of any active component.

The staff finds this response to be acceptable, and that the applicant's containment leak-test program will meet the criteria of Appendix J. Confirmatory Issue (24), listed in Section 1.10 of the SER, is accordingly resolved.

6.2.7 TMI-2 Requirements

(3) Containment Isolation Dependability (TMI Action Plan Item II.E.4.2)

Discussion and Conclusion

In Section 6.2.7(3) of the SER, the staff evaluated the applicant's compliance with the TMI Action Plan Item II.E.4.2 (NUREG-0737) requirements and identified, as a confirmatory issue, the need for the applicant to classify essential and nonessential systems in regard to the containment isolation system - Item (b) - and to specify the minimum containment pressure setpoint that will be compatible with normal operating conditions - Item (e). The applicant provided additional information in a letter dated June 8, 1982 (D. R. Davidson to A. Schwencer), which acceptably classified the systems penetrating containment into essential and nonessential systems, indicating the essential systems to be engineered safety features systems that are required for accident situations or shutdown, and which stated that the minimum containment pressure setpoint, still being determined, would be provided and reviewed in connection with the preparation of the NRC Technical Specification for Perry.

The staff considers this response to be acceptable to fully meet the requirements of TMI Action Plan Item II.E.4.2 and that Confirmatory Issue (23), listed in Section 1.10 of the SER, is resolved.

6.3 Emergency Core Cooling System (ECCS)

6.3.1 System Description

6.3.1.3 Functional Design

Requirement

In Section 6.3.1.3 of the SER, the staff reported, as part of its review and evaluation of the Perry ECCS functional design, that the automatic depressurization system (ADS) CEI is planning to incorporate can be used as a backup to the high-pressure cooling systems and allows the functioning of the low-pressure cooling systems in the event of a small-line break. The air supplied to the ADS valves will be provided in accident conditions by seismically qualified accumulators and receivers to compensate for leakage past accumulator check valves in accordance with TMI Action Plan Item II.K.3.28 requirements. Air or nitrogen accumulators for the ADS valves are provided with sufficient capacity to cycle the valves open five times at design pressures. The General Electric Company has also stated that the ECCS is designed to withstand a hostile environment and still perform its function within 100 days following an accident.

Discussion

The staff required that the applicant demonstrate that the ADS valves, accumulators, and asssociated equipment and instrumentation meet the requirements of TMI Action Plan Item II.K.3.28 and are capable of performing their intended functions during and following exposure to hostile environments, taking no credit for nonsafety-related equipment or instrumentation. Additionally, air (or nitrogen) leakage through valves must be accounted for to ensure that enough inventory of compressed air is available to cycle the ADS valves. If this cannot be demonstrated, it must be shown that the accummulator design is still acceptable.

The applicant's commitment to satisfy the requirements of TMI Action Plan Item II.K.3.28 is discussed in the Licensing Review Group-II (a group formed to address BWR/6 issues on a generic basis) position paper submitted by letter from D. L. Holtzscher (Illinois Power Company) to H. J. Faulkner (NRC) dated May 17, 1982. The ADS design described in the position paper is identified as Generic Issue 8-RSB. In a letter dated September 16, 1982 (D. R. Davidson to A. Schwencer), CEI as a member of the Licensing Review Group-II (LRG-II) adopted the LRG-II position regarding the ADS design for Perry, which is described below.

Design Description

The ADS design described in the LRG-II position paper, and adopted for Perry by CEI, uses selected safety/relief valves (SRVs) for depressurization of the reactor. Each SRV utilized for automatic depressurization is equipped with an air accumulator, a check valve, and a safety-grade backup air supply. The safety-grade ADS pneumatic supply is separated into two divisions. The loss of air supply to one division of ADS valves will not prevent the ability of the ADS to depressurize the reactor system if it is required. The ADS accumulators are designed to provide two SRV actuations at 70% of drywell design pressure, which is equivalent to four actuations at atmospheric pressure. Normal air is supplied for the ADS valve accumulator from an air compressor located in the Perry auxiliary building. This compressor supplies air to two inline air receiver tanks located in the Perry intermediate building, each having a volume of 10.5 ft^3. The compressor automatically maintains air pressure in these two tanks between 2,250 and 2,500 psig. Each tank serves one division of ADS valve accumulators by means of a 2,500/150-psig pressure regulating valve. If the air compressor is not available, the compressed air in the two inline receiver tanks serves as the backup air supply and can recharge the ADS valve accumulators to provide makeup for any system leakage for a period of 7 days. Both tanks have a connection on downstream piping to permit commercially available air or nitrogen bottles to be connected to the system to ensure a 100-day postaccident ADS air supply.

The ADS air supply system, from the two inline check valves located upstream of each air receiver tank to the valve accumulators, is designed to the requirements of ASME Code, Section III, Class 3, and is seismic Category I. The section of this line penetrating the containment and the inboard and outboard isolation valves are designed to the requirements of ASME Code, Section III, Class 2, and are seismic Category I.

In the event of a loss of air supply from the air compressor, one or more of the following control room alarms would be activated:

(1) receiver tank air pressure low (2,000 psig)
(2) air compressor/purifier package inoperable

When the alarm in the control room indicates low receiver tank pressure, the air compressor is manually started and runs until the system pressure is returned to the normal operating range. When the alarm indicates the compressor is inoperable, receiver tank pressure is monitored while the compressor problem is evaluated. If the compressor cannot be restarted in a timely fashion, then commercially available air or nitrogen bottles can be connected to the safety-class connections near the air receiver tank to supplement the tank's supply during repairs. The connections for the air or nitrogen bottles are located outside the reactor building in the auxiliary building or the control complex and are accessible in the event of an accident.

Evaluation

The staff has evaluated the LRG-II position paper design adopted for Perry by CEI and finds that the ADS backup air supply system has been designed for sufficient inventory to cycle the ADS valves in the event they are required to operate. In addition, the large receivers will be monitored in the control room to ensure there is sufficient inventory of air to cycle the ADS valves according to design requirements. If the air receivers' inventory drops below 2,000 psig, and the air compressor is not available to restore the receiver pressure to between 2,250 and 2,500 psig, additional air can be provided by remote bottle hookups.

The following surveillance requirements will be incorporated into the NRC Technical Specifications to ensure the Perry backup air system will provide continued long-term assurance of the availability of sufficient inventory of air to actuate the ADS valves if they are needed:

(1) At least once every 31 days, perform a channel functional test of the accumulator backup compressed gas system low-pressure alarm systems.

(2) At least once every 18 months, perform a channel calibration of the accumulator backup compressed gas system low-pressure alarm systems and verify the air alarm setpoint of 2,000 $\pm$ 75 psig on decreasing pressure.

This periodic surveillance of the low-pressure alarm system, in conjunction with the design of the backup air supply systems to the ADS valves, should ensure an adequate and operable backup air supply to the ADS valves in the event they are required to operate. The staff, therefore, concludes that the LRG-II position design adopted for Perry by CEI is acceptable and will meet the requirements of TMI Action Plan Item II.K.3.28.

6.4 Control Room Habitability Systems

In Section 6.4 of the SER, the staff considered control room habitability acceptable for meeting TMI Action Plan Item III.D.3.4 and GDC 19. However, in a separate review of the control room fire protection systems, as reported in Section 9.5.1.6.2 of the SER, the staff was concerned that the use of a carbon dioxide (CO_2) fire extinguishing system in the underfloor spaces of the control room could jeopardize the habitability of the control room. In a

letter (D. R. Davidson to A. Schwencer) dated August 31, 1982, the applicant provided further details of the CO_2 system and the consequences of its use. The fire extinguishing sytem is designed to discharge CO_2 in short bursts into any of three subfloor zones where there is a fire and repeat this process if reignition occurs.

The staff has reviewed the potential consequences of the eventual entry of CO_2 and other gases produced by combustion into the control room atmosphere, and, in light of the additional information furnished by the applicant, has concluded that these gases do not jeopardize the habitability of the control room. In the case of a lingering fire requiring multiple discharges from the system, discharges over tens of minutes would be required before unhealthy concentrations of CO_2 and pyrolysis gases would accumulate in the control room. Because the system automatically alarms within the control room, adequate time is available for personnel to don self-contained breathing apparatuses supplied for this purpose. The total amount of CO_2 (1,000 lb) is capable of displacing approximately 3% of the 268,000-ft^3 free volume of the control room envelope. The current permissible 8-hour shift exposure to CO_2 is 0.5% (as specified in the Occupational Safety and Health Act (OSHA) National Institute of Occupational Safety and Health (NIOSH) Publication 81-123, 1981), which was adopted from the existing industry standard. NIOSH has recommended limits of 1% CO_2 exposure averaged over 40 hours per week and 3% for routine exposures of less than 10-minutes duration. The amount of CO_2 in standard air is 0.04%, in urban air it is often as high as 0.1%, and in human exhalation it is approximately 5%. Self-contained breathing apparatuses are required by OSHA for work places having concentrations of 5% or more. Concentrations as high as 1.5% are permitted in submarines and space craft (American Industrial Hygienists Association, <u>Hygienic Guide</u>). Should the entire CO_2 inventory of the system be discharged into the control room envelope while it is isolated from outside air, it is recommended that the operators don self-contained breathing apparatuses to reduce the likelihood of minor effects such as labored breathing or delayed headaches, but such action would not be required to prevent incapacitation. The control room smoke purge system has the capacity, if manually initiated, for exchanging 10% per minute of the control room air volume with the outside atmosphere, or reducing excess CO_2 concentrations by half every 7 minutes.

As a consequence of the above considerations, the staff concludes that use of the CO_2 fire extinguishing system is acceptable with respect to control room habitability. This evaluation finding, together with that presented in Section 9.5.1.6.2 of this supplement, resolves Outstanding Issue (16) listed in Section 1.9 of the SER.

7 INSTRUMENTATION AND CONTROLS

7.2 Reactor Protection Systems

7.2.2 Specific Findings

7.2.2.4 Scram Discharge Volume Level Monitoring System

The staff reported in the SER that, as a result of the Browns Ferry event, where a complete insertion of the control rods was not successful until several attempts had been made, the applicant had modified the design of the scram discharge instrument volume level monitoring system to preclude such an event occurring in Perry. Although the staff found the modified design acceptable, the applicant was required, as a confirmatory issue, to provide a complete description of the modified design in the FSAR. In a letter dated October 25, 1982 (D. R. Davidson to A. Schwencer), the applicant provided documentation to be included in a future FSAR amendment that the staff finds acceptably describes the modified design approval. Therefore, Confirmatory Issue (27), listed in Section 1.10 of the SER, is resolved.

7.3 Engineered Safety Features (ESF) Systems

7.3.2 Specific Findings

7.3.2.2 High-Pressure Core Spray System

In the SER the staff reported its evaluation of the Perry high-pressure core spray (HPCS) system initiation circuitry design including its concurrence with the applicant's argument that a high drywell pressure interlock, which the staff had found was needed to prevent premature termination of HPCS, be excluded. (The applicant had maintained that the interlock would tend to keep HPCS in operation past the point necessary to reflood the core, causing steamline flooding, and that addition of the interlock would not significantly increase the overall safety of the plant.) Although the staff found the design, excluding the interlock, acceptable, it required the applicant, as a confirmatory issue, to formally submit the revised system design for review and to incorporate the revised design in the plant before start of operation. The applicant provided the final design approved by the staff in a letter dated October 14, 1982 and committed to incorporate the design before start of plant operation. The staff has confirmed that the final design provided in that letter is acceptable, thus resolving Confirmatory Issue (28), listed in Section 1.10 of the SER. A confirmatory site audit of the HPCS initiation circuitry design will be conducted before fuel load of Unit 1 to verify that the final design acceptable to the staff has been installed.

7.3.2.6 Periodic Testing of ESF Actuation Systems During Plant Operation

During the staff's review of the capability to test the pilot solenoid valves that control compressed air to the automatic depressurization system (ADS)

relief valves, it became apparent that the present Perry design does not provide a feature to test the solenoid valves and associated circuitry with the plant at power. In the GESSAR-238 nuclear steam supply system preliminary design SER (Docket No. STN 50-550), dated March 1977, the staff identified this as a potential problem and took the position that GE would be required to make provisions to improve the testability of the ADS solenoid valves during reactor operation.

The staff continued to pursue with the applicant the adequacy of the ESF actuation system design from the standpoint of providing the capability to periodically test the actuation circuits with the plant at power. In a letter from the applicant dated April 29, 1982 (D. R. Davidson to A. Schwencer) and in a subsequent meeting, this issue was discussed. The applicant described the capability of testing ESF systems at power. For example, it is possible to test portions of the circuitry to energize a pump while maintaining the injection valve closed and subsequently test the injection valve while maintaining the pump deenergized.

This capability exists for all ESF systems except the ADS. The ADS is unique in that the ADS solenoid valves cannot be tested without causing the safety/relief valves to open. The applicant stated that the ADS pilot solenoid operability test is performed at least once every 18 months at a reduced reactor steam dome pressure. GE has investigated the benefit of adding valves to test the ADS pilot solenoid valve at power and has concluded (letter, G. G. Sherwood to B. C. Rusche, August 9, 1976) that those modifications that would permit full ADS solenoid valve testing would decrease the reliability of the ADS. It should be noted that redundant solenoid valves operated from redundant divisions are provided for each ADS valve. The applicant has also investigated continuity checks of the solenoid valve at power and has stated that there is no reason to believe and no experience to indicate that continuity of the solenoid valve is questionable while the rest of the ADS is operable. Furthermore, the applicant states that given the proven reliability of the solenoid valve, the weakest link is more probably the mechanical motion of the solenoid plunger, which the continuity check would not verify. As a result of GE's and the applicant's investigations, it is apparent to the staff that to modify the current ADS design to include an additional continuity check of the solenoid valve will only slightly improve the proven reliability of the ADS pilot solenoid valve. Therefore, the staff has concluded that the addition of a continuity check for the ADS solenoid valves would not improve the reliability of the ADS sufficiently to justify the complexity of the required check.

In addition, the staff has concluded that the current surveillance test interval for the ADS pilot solenoid valve (at least once every 18 months) and the monthly ADS logic testing, which tests through the relay contacts in the valve solenoid circuit and causes a panel light to come on indicating proper channel operation, are sufficient. These intervals for testing have been established and approved by the staff for operating BWRs. This resolves Outstanding Issue (11) listed in Section 1.9 of the SER.

7.3.2.7 Manual Initiation and Termination of ESF Systems

During its review of the ESF systems, the staff found that the logic for manual initiation for several ESF systems is interlocked with permissive logic from

various sensors. In some cases it appears that the permissive logic is dependent on the same sensors as those used for automatic initiation of the system. The staff questions whether this design meets the intent of Institute of Electrical and Electronics Engineers Standard 279, Section 4.17.

The staff still considers Outstanding Issue (12) in Section 1.9 of the SER to be unresolved and is continuing to pursue with the applicant the adequacy of the design for manually initiating and terminating safety systems at the system level. In a letter dated July 20, 1982 (D. R. Davidson to A. Schwencer), the applicant provided additional information that lists the ESF systems in which the logic for manual initiation is interlocked with permissive logic from various sensors. The applicant has also provided justification for the presence of these permissive interlocks. The staff is currently reviewing this information and will report its findings on this issue in a future supplement to the SER.

7.4 Systems Required for Safe Shutdown

7.4.2 Specific Findings

7.4.2.2 Remote Shutdown System

The applicant had indicated that several of the readouts and associated sensors and power supplies on the remote shutdown panel were not safety grade. The applicant has reviewed the design to determine whether these nonsafety-grade readouts and associated sensors and power supplies are required to achieve shutdown. The results of this review (letter dated April 29, 1982 from D. R. Davidson to A. Schwencer) have indicated that all of the readouts and associated sensors and power supplies that are required to achieve shutdown are safety grade. The staff concludes that this confirmatory issue is resolved and that the design of the remote shutdown system is in conformance with the applicable criteria and is, therefore, acceptable. This resolves Confirmatory Issue (29) listed in Section 1.10 of the SER.

7.4.2.4 RCIC Test Procedures

On the basis of its review of the reactor core isolation cooling (RCIC) test procedures furnished by the applicant, and reported in Section 7.4.2.4 of the SER, the staff found that they demonstrated that the RCIC system can be tested adequately during plant operation and were therefore acceptable. However, the staff noted several minor discrepancies between the RCIC system logic, as shown on an electrical-elementary diagram (B-208-075, System E51), and the test procedures, which the staff required the applicant to correct as a confirmatory issue. In a letter dated October 14, 1982, the applicant confirmed that the discrepancies were evident and committed to correct all of the diagrams to be consistent with the procedures accepted by the staff. The staff finds that the applicant's commitment satisfactorily addresses this issue. Therefore, Confirmatory Issue (30), listed in Section 1.10 of the SER, is resolved.

7.5 Safety-Related Display Instrumentation

7.5.2 Specific Findings

7.5.2.5 Additional Accident-Monitoring Instrumentation (TMI Action Plan Item II.F.1, Positions 4, 5, and 6)

In reviewing the applicant's conformance with the requirements of TMI Action Plan Item II.F.1, Positions 4, 5, and 6 (II.F.1.4, II.F.1.5, and II.F.1.6), the staff found that the containment pressure monitor, the containment water level (suppression pool) monitor, and the containment hydrogen monitor, described in the applicant's letter dated April 20, 1982, appropriately provide redundant, seismically and environmentally qualified instrumentation, powered from reliable power sources, and found to be of suitable range. However, the staff required, as a confirmatory issue, that the applicant define the accuracy of the monitors to be used at Perry to determine full conformance with TMI Action Plan Items II.F.1.4, II.F.1.5, and II.F.1.6. The applicant provided this information in a letter dated October 14, 1982. The accuracy of the monitors is considered to be acceptable to the staff. Therefore, Confirmatory Issue (32), listed in Section 1.10 of the SER, is resolved.

7.7 Control Systems

7.7.2 Specific Findings

7.7.2.3 Failures in Vessel Level Sensing Lines Common to Control and Protection Systems (LRG-II Generic Issue 1-ICSB)

In Section 7.7.2.3 of the SER, the staff reported its findings regarding an analysis performed by the Licensing Review Group-II (LRG-II), and adopted for Perry by CEI, which generically addresses a break in a level sensing line common to reactor control and protection systems. The LRG-II analysis considered a combination of the worst-possible failure in a protective channel. The staff concluded that although the LRG-II analysis was predicated on a 251-in.-core-diameter BWR/6 plant, the results were applicable to the Perry 238-in. core diameter because the plant size characteristics analyzed were found to be conservative, in that the water level in the core would not be lower for Perry and that the scenario postulated in the LRG-II analysis would have no adverse safety consequences. However, in describing the applicability of the LRG-II findings to Perry, the applicant indicated that when the recirculation pumps trip off, they would be transferred to the low frequency motor generator sets, which the staff found to be inconsistent with the LRG-II analytical results. The applicant was required to confirm the basis for this inconsistency as a confirmatory issue. In a letter dated October 14, 1982 (D. R. Davidson to A. Schwencer), the applicant responded that it was in error and that the pumps would not transfer to the motor generator sets after tripping off. The staff considers this response acceptable and consistent with the LRG-II findings. Thus, Confirmatory Issue (33), listed in Section 1.10 of the SER, is resolved.

8 ELECTRIC POWER SYSTEMS

8.2 Offsite Power Systems

8.2.4 Adequacy of Station Electric Distribution System Voltages

In Section 8.2.4 of the SER, the staff reported the results of its evaluation of the Perry design for conformance with Branch Technical Position (BTP) PSB-1, "Adequacy of Station Electric Distribution System Voltages." The staff required, as a confirmatory issue, that the applicant provide the final design of the first and second level undervoltage protection of the safety equipment in accordance with BTP PSB-1 before plant startup.

The applicant provided details of the final design in a letter dated August 26, 1982 (D. R. Davidson to A. Schwencer). The design provides three redundant and independent emergency buses, each having two levels of undervoltage protection: (1) loss of power and (2) degraded grid voltage. The loss of power protection at the 4.16-kV emergency buses consists of two sets of three single-phase instantaneous undervoltage relays with a setpoint at 75% of the equipment rated voltage. The relays are arranged in a two-out-of-two coincident logic to initiate a timer with a 3-sec time delay. In the event that voltage loss is maintained for 3 sec, the timer trips the offsite power source breakers and will initiate the necessary logic to start the diesel generators and connect the Class 1E buses to the diesel generators.

The degraded grid voltage protection at the 4.16-kV emergency buses consists of two sets of three single-phase instantaneous undervoltage relays with a setpoint at 96% of the equipment rated voltage. The relays are arranged in a two-out-of-two coincident logic to initiate two separate time-delay relays. The first time-delay relay will initiate undervoltage alarms after 15 sec. If an undervoltage condition between 96% and 75% occurs, concurrent with a LOCA, the emergency bus is retained on offsite power for a 15-sec time delay, after which the offsite source breakers will be tripped and the diesel generator will supply power to Class 1E loads through the load sequencer. The second time delay will be set for 5 minutes at which time the offsite source breakers will be tripped and the diesel generator will energize the emergency bus. The Class 1E equipment (motors) is capable of starting and accelerating its specified load with 75% of rated voltage at the equipment terminals. The 5-minute timer will also ensure against the motors' overheating between 90% and 75% voltage.

The staff finds that the applicant's final design of the first and second level undervoltage protection of safety equipment is acceptable; thus, Confirmatory Issue (34), listed in Section 1.10 of the SER, is satisfactorily resolved.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage Facility

9.1.5 Overhead Heavy-Load-Handling System

In Section 9.1.5 of the SER, the staff stated that the applicant had committed to implement the interim actions before the final implementation of the NUREG-0612 guidelines and before the operating license is issued. Additionally, the applicant, by letter dated September 15, 1982, committed to full compliance with the guidelines of NUREG-0612. The applicant has made three submittals dated June 19, 1981, June 9, 1982, and September 15, 1982, concerning the implementa'ion of Phase I of NUREG-0612. The staff's review of the applicant's submittals is continuing. However, the staff will require that a condition be placed in the license requiring that 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 the NRC generic letter dated December 22, 1980). Before startup following the second refueling outage, the applicant shall have made commitments acceptable to the NRC regarding the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II - 9-month responses to the NRC generic letter dated December 22, 1980).

9.3 Process Auxiliaries

9.3.4 Standby Liquid Control System

In a letter dated August 13, 1982 (D. R. Davidson to A. Schwencer), the applicant summarized a change in the design of the standby liquid control system that includes an increased flow capacity from 43 gpm to 86 gpm. This will involve increasing the size of both pumps' suction lines as well as changing the reactor vessel injection point to the high-pressure core spray system header. Although the design will include both manual and automatic initiation capability, only manual initiation will be functional. Details of the design change will be documented by the applicant in a future amendment to the applicant's FSAR.

The planned design change is still considered to be an outstanding issue pending staff review of the final design details. The staff's evaluation findings on this issue will be addressed in a future supplement to the SER.

9.5 Fire Protection Systems

9.5.1 Introduction

9.5.1.4 General Plant Guidelines

9.5.1.4.2 Safe Shutdown Capability

In Section 9.5.1.4.2 of the SER, the staff considered, as an outstanding issue, the need for the applicant to provide a satisfactory analysis of the plant fire

protection features that ensure safe shutdown capability in accordance with Branch Technical Position CMEB 9.5-1, Section C.6.b (10 CFR 50, Appendix R, Section III G). In a letter dated June 16, 1982 (D. R. Davidson to A. Schwencer), the applicant addressed this issue and identified several plant areas that deviate from the equipment separation requirements of Appendix R, Section III G. The staff has reviewed this information and finds (1) that the analysis performed by the applicant will ensure that the plant fire protection measures will meet requirements and (2) that protection measures to be taken in those areas that deviate from the provisions of Appendix R, Section III G, are acceptable in protecting equipment needed for safe shutdown. The applicant was advised of the resolution of this issue (identified as Outstanding Issue (15) listed in Section 1.9 of the SER) in a letter from A. Schwencer to D. R. Davidson dated November 26, 1982. A more comprehensive documentation of the staff's findings and basis for accepting the proposed deviations to Appendix R will be provided in a future supplement to the SER.

9.5.1.6 Fire Protection of Specific Plant Areas

9.5.1.6.2 Control Room

In Section 9.5.1.6.2 of the SER, the staff considered, as an outstanding issue, the use of carbon dioxide (CO_2) as a fire suppression agent in the control room. The staff reported that (1) CO_2 had not been tested and approved as a fire suppressant in the GE power generation control complex (PGCC) system selected for installation in the plant; (2) GE Topical Report NEDO-10466, Revision 2, dated March 1978, which pertains to the selected PGCC system, and approved by the staff, specifies that Halon 1301 fire protection is required; and (3) there were concerns that CO_2 might leak from the PGCC underfloor into the control room (e.g., because of an inadvertent activation of the system) resulting in possible injury to the operators and the forced evacuation of the control room. In response to an appeal from the applicant for the staff to reconsider its objection to the CO_2 system, the staff's specific concerns were detailed in a letter dated June 9, 1982 (A. Schwencer to D. R. Davidson). In a letter dated August 31, 1982, the applicant addressed each of the staff's concerns and presented details of planned system design modifications that reduce the amount of CO_2 that could be hazardous to the operators, while at the same time providing effective fire suppressant capability.

The staff has reviewed the applicants' responses and evaluated the planned design modifications; it finds that its concerns regarding the use of CO_2 have been satisfactorily addressed, and that the modified system design will preclude impairment of the operator's ability to maintain safe shutdown conditions from the control room in the event of a fire or because of an inadvertent discharge of CO_2. The resolution of this issue, identified as Outstanding Issue (16) in Section 1.9 of the SER, was communicated to the applicant in a letter from A. Schwencer to D. R. Davidson dated November 26, 1982. A more detailed and comprehensive documentation of the staff's fire-protection review findings and basis for accepting the CO_2 system instead of the required Halon 1301 system in the control room PGCC system will be provided in a future supplement to the SER.

9.6 <u>Other Auxiliary Systems</u>

9.6.3 Emergency Diesel Engine Fuel Oil Storage and Transfer System

9.6.3.1 Emergency Diesel Engine Auxiliary Support Structures (General)

(4) Testing No-Load and Light-Load Operation

In Section 8.3.1 of the FSAR, the applicant discussed the manufacturer's recommendations for no-load and light-load operation of the diesel generators. In a letter dated March 25, 1982, the applicant committed to implement the following procedures before startup:

(a) Whenever a diesel is started and operated for an extended period of time (that is, not terminated within 2 minutes), the diesel generator shall be loaded to at least 25% of full load for a minimum of 30 minutes.

(b) During periodic testing, the diesel will be loaded to a minimum of 25% of full load or as recommended by the manufacturer.

(c) During troubleshooting, no-load operation will be minimized. If the troubleshooting operation extends over a period of time (that is, 3 to 4 hours or more), the engine shall be cleared in accordance with the manufacturer's recommendations for no-load and light-load operation.

On the basis of its review of the diesel generator testing no-load and light-load operation and procedures to be implemented by the applicant, the staff concluded in Section 9.6.3.1 of the SER that the applicant needed to define "extended period of time" of operation and when, during this time period of no-load and light-load operation, the diesel generator would be electrically loaded. The staff also required that Procedure (a), described in the applicant's letter dated March 25, 1982 and listed above, be modified as follows: Whenever the diesel generator is started and its operation is not terminated within 2 minutes after startup, the diesel generator shall be loaded to at least 25% of full load for a minimum of 30 minutes; if the diesel generator is operated in a no-load or light-load mode for an extended period of time (not to exceed 8 hours), the generator shall be loaded to a minimum of 25% of full load for a minimum of 30 minutes, or as recommended by the manufacturer. Finally, that Procedure (a) as modified and Procedures (b) and (c) listed above are to be implemented by the applicant before plant startup.

In Section 8.3.1.1.3.2 of the FSAR, Amendment 8 (CEI letter dated March 25, 1982), the applicant documented his commitment to implement Procedure (a), modified as required by the staff, and Procedures (b) and c) for diesel generator no-load and light-load operation and defined "extended period of time" for such operation. This response is acceptable to the staff and satisfactorily resolves Confirmatory Issue (50), listed in Section 1.10 of the SER - this issue was initially listed as License Condition (8) in the SER, but was changed to Confirmatory Issue (50) in Supplement No. 1 to the SER.

12 RADIATION PROTECTION

12.3 Radiation Protection Design Features

12.3.2 Shielding

Section 12.3.2 of the SER contained a conditional acceptance of the shielding for the inclined fuel transfer system (IFTS), subject to (1) the applicant providing plant drawings of areas through which spent fuel bundles pass and (2) performance of a shielding design review demonstrating that radiation zones meet the staff positon stated in Section 12.3 of NUREG-0800. The necessary information was provided in Amendment 10 to the Perry FSAR and referenced in the applicant's letter (M. R. Edelman to B. J. Youngblood) dated December 21, 1982.

The results of the applicant's shielding design review of the IFTS demonstrate that accessible areas through which spent fuel bundles pass will either meet the maximum instantaneous dose rate permitted in NUREG-0800 or will be controlled using an electronic key activation system. The electronic key system will not allow access to an area during spent fuel transfer.

The staff accordingly concludes that the Perry plant's shielding for the IFTS meets the criteria of NUREG-0800 and is therefore acceptable. This action resolves Confirmatory Issue (44), listed in Section 1.10 of the SER.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

In Section 12.3.4.1 of the SER, the staff found the applicant's containment high-range gamma monitoring system acceptable. However, the staff required plant layout drawings identifying the location of the monitors to complete its evaluation and, therefore, considered this to be a confirmatory issue. In a letter dated October 14, 1982 (D. R. Davidson to A. Schwencer), the applicant provided specific information on the location of the four area radiation monitors instead of providing plant layout drawings. The staff finds that this additional information adequately addresses this issue in that the postaccident high-range monitors will be located in the plant in compliance with TMI Action Plan Item II.F.1.3 requirements. Therefore, the staff concludes that Confirmatory Issue (45), listed in Section 1.10 of the SER, has been satisfactorily addressed by the applicant and is considered to be resolved.

13 CONDUCT OF OPERATIONS

13.2 Training Program

13.2.1 Licensed Operator Training Program

13.2.1.9 Operator Requalification Program

In Section 13.2.1.9 of the SER, the staff found as a confirmatory issue that the applicant's operator training program required further emphasis on teaching plant operators the use of equipment and systems to control or mitigate accidents in which the core is severely damaged to fully comply with the requirements of TMI Action Plan Item II.B.4. In a letter dated September 13, 1982 (D. R. Davidson to A. Schwencer), the applicant furnished an outline of the program under development that the staff finds adequately addresses this confirmatory issue. The applicant's response has also been found to be consistent with the criteria in Section 13.2.1 of the Standard Review Plan (NUREG-0800). Thus, Confirmatory Issue (46), listed in Section 1.10 of the SER, is considered to be resolved.

13.2.2 Training for Nonlicensed Plant Staff

13.2.2.1 Conclusion

In Section 13.2.2.1 of the SER, the staff reviewed the applicant's overall training program for nonlicensed plant staff and concluded that the training program met the requirements of American National Standards Institute (ANSI) N18.1-1971 as endorsed by Regulatory Guide 1.8, Revision 1, and 10 CFR 50, Appendix R, Section I, except that, as a confirmatory issue, the applicant was required to furnish additional information on the training program to be established for the Nuclear Project Training Section (NPTS). In a letter dated September 13, 1982 (D. R. Davidson to A. Schwencer), the applicant provided additional information in response to this confirmatory issue.

On the basis of its review of the additional information provided by the applicant and a site visit made by the staff on November 2, 1982, which consisted of interviews with selected CEI project personnel, the staff found that the NPTS is developing procedures, programs, and recordkeeping activities to allow the Perry Plant Department Training Unit (PPDTU) to be absorbed into the NPTS at fuel load of Unit 1. The PPDTU is working with the NPTS in this development and in the consolidation of training of construction and operating engineering personnel. BWR systems training for construction and operating engineering personnel has been provided by the PPDTU. Similar training is planned for Nuclear Test Department personnel. Training of nonlicensed personnel is being removed from these departments and assigned to the PPDTU. The PPDTU is developing qualification standards for all Perry plant department positions for which training is required. These standards will be based on job specification, guidelines of the Institute of Nuclear Power Operations (INPO), industry experience, and department input. INPO generic task analysis will also be used in

this development effort. The present technical personnel staffing levels are 85 in the NPTS and 12 in the PPDTU with projections at Unit 1 fuel load of 15 for the NPTS and 17 for the PPDTU.

The staff concludes that the applicant's training program, when it is reorganized and the staff is complete, will meet the applicable requirements stated above and is consistent with the acceptance criteria of NUREG-0800, Section 13.2.2. Therefore, Confirmatory Issue (47), listed in Section 1.10 of the SER, is considered to be satisfactorily resolved.

13.5 Plant Procedures

13.5.1 Administrative Procedures

13.5.1.8 Shift Supervisor Responsibilities

In its review of the applicant's administrative procedures, as reported in Section 13.5.1.8 of the SER, the staff found the duties, responsibilities, and authority of the Shift Supervisor and control room operators to be a confirmatory issue and required the applicant to provide a written commitment to have the Shift Supervisor training program emphasize and reinforce the management functions and authority of the Shift Supervisor to ensure safe operation of the plant in conformance with the requirements of TMI Action Plan Item I.C.3.

In a letter dated September 13, 1982 (D. R. Davidson to A. Schwencer), the applicant provided the following written commitment in response to this confirmatory issue:

> A corporate management directive will be issued establishing the command duties of the Shift Supervisor that emphasizes the primary management responsibility for safe operation of the plant. Plant administrative procedures will define the duties, responsibilities and authority of the Shift Supervisor and control room operators. The Shift Supervisor training program will emphasize and reinforce the responsibility for safe operation of the plant and the Shift Supervisor's role in assuring plant safety.

The staff finds that this written commitment is acceptable in meeting the training requirements of TMI Action Plan Item I.C.3 and that it conforms with the criteria of NUREG-0800, Section 13.5.1. Thus, Confirmatory Issue (48), listed in Section 1.10 of the SER, is resolved.

13.5.1.11 Verify Correct Performance of Operating Activities

In SER Section 13.5.1.11, the staff evaluated the applicant's procedure for equipment control and surveillance audit activities to verify correct performance of the plant operating activities for compliance with TMI Action Plan Item I.C.6 requirements. The staff concluded this area to be a confirmatory issue subject to receipt of a description of the applicant's plans for verifying correct implementation of equipment control measures in radiation areas. In a letter dated September 13, 1982 (D. R. Davidson to A. Schwencer), the applicant stated that procedures for verifying the functional acceptability of any equipment that is important to safety are under development and will be fully in

accordance with TMI Action Plan Item I.C.6. For equipment control measures in radiation areas, as low as is reasonably achievable considerations will be taken into account in the procedure development.

The staff considers this response acceptable and finds that the procedures to be developed will meet the requirements of TMI Action Plan Item I.C.6 and the criteria of NUREG-0800, Section 13.5.1. Therefore, Confirmatory Issue (49), listed in Section 1.10 of the SER, is resolved.

16 TECHNICAL SPECIFICATIONS

On the basis of the staff's review of the Perry emergency core cooling system's functional design regarding the adoption by CEI of the Licensing Review Group-II position paper on the automatic depressurization system (ADS) design for Perry (discussed in Section 6.3.1.3 of this supplement), Item (7) is being added to those listed in Section 16 of the SER to require the applicant to perform a periodic surveillance of the low-pressure air alarm systems as an NRC Technical Specification. This surveillance requirement is being imposed to ensure that the Perry backup air system will provide continued long-term assurance of the availability of a sufficient inventory of air to actuate the ADS valves if they are needed. Specifically, at least once every 31 days, the applicant will be required to perform a channel functional test of the accumulator backup compressed air system low-pressure alarm systems, and at least every 18 months, the applicant will be required to perform a channel calibration of the accumulator backup compressed gas system low-pressure alarm systems and verify the air alarm setpoint of 2,000 ± 75 psig on decreasing pressure.

A composite list of the items contained in Section 16 of the SER, including the item discussed above, follows (the SER section in which each item is discussed is given in parentheses):

(1) Response-time testing (7.2.2.6)

(2) Systems shared by Units 1 and 2 (7.3.2.4)

(3) Single-loop operation subject to staff approval of supporting analysis (4.4.4)

(4) Operation in natural circulation made subject to completion of staff generic evaluation of thermal-hydraulic stability for BWRs (4.4.4)

(5) Core flow is to be checked by the applicant at least once every 24 hours to account for possible buildup of crud deposition (4.4.5)

(6) Availability, setpoints, and frequency of surveillance requirements will be imposed for Level 8 trip and turbine bypass equipment to ensure an acceptable level of performance for anticipated operational transients (15.1)

(7) Periodic surveillance of the low-pressure air alarm systems to ensure that the backup air system will provide continued inventory of air to actuate the ADS valves if they are needed (6.3.1.3)

18 CONTROL ROOM DESIGN REVIEW

The staff reviewed the applicant's interim report describing the detailed control room design review (DCRDR) for Unit 1, submitted by CEI letter dated June 7, 1982 (D. R. Davidson to A. Schwencer), and has performed an in-progress audit of the Unit 1 control room in August 1982. A report of the staff's audit findings were submitted to CEI in a letter dated November 12, 1982. Safety significant human engineering deficiencies (HEDs) identified by the applicant in the DCRDR and those found during the staff's in-progress audit are to be corrected before Unit 1 is licensed. If a safety significant HED cannot be corrected before licensing, the applicant will be required to provide a rationale for deferral, an interim prelicensing correction, and a reasonable schedule for implementing the longer-range correction. The staff will assess the extent to which HEDs, which are to be corrected subsequent to licensing, should be cited as a condition in the operating license issued, including a time when each particular HED must be resolved during Unit 1 operation.

CEI has been asked to arrange a schedule for the staff's final audit, which is to address the closeout of TMI Action Plan Items I.D.1 and II.K.3.27. Until this final audit is satisfactorily completed, Outstanding Issue (7), listed in Section 1.9 of the SER, will continue to remain unresolved.

APPENDIX A

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

NOTE: This appendix lists correspondence between NRC and the applicant inadvertently omitted in the SER and Supplement No. 1, in addition to continuing the chronology, in this supplement.

March 13, 1981	Letter from applicant submitting Amendment 1 to the FSAR.
May 22, 1981	Letter from applicant submitting Amendment 2 to the FSAR.
June 19, 1981	Letter from applicant transmitting Gilbert Associates report on control of heavy loads (NUREG-0612).
September 11, 1981	Letter from applicant submitting Amendment 3 to the FSAR.
October 2, 1981	Letter from applicant submitting Amendment 4 to the FSAR.
November 3, 1981	Letter from applicant submitting Amendment 5 to the FSAR.
November 3, 1981	NRC letter to applicant regarding NUREG-0737, Item II.K.3.44.
February 10, 1982	Letter from applicant submitting Amendment 6 to the FSAR.
March 25, 1982	Letter from applicant responding to SER Confirmatory Issue (50).
May 6, 1982	Letter from applicant responding to NRC staff concerns regarding fire protection issues.
May 20, 1982	Letter from applicant responding to draft SER structural engineering concerns.
May 27, 1982	Letter from applicant submitting Amendment 7 to the FSAR.
June 2, 1982	Letter from applicant responding to questions concerning the structural design of the Perry intake and discharge water tunnels.
June 2, 1982	Letter from applicant providing the plan for preoperational vibration monitoring program for balance-of-plant systems.
June 2, 1982	Letter from applicant documenting May 12, 1982 equipment qualification meeting.
June 7, 1982	Letter from applicant responding to request for additional information regarding containment buckling analyses.

June 7, 1982	Letter from applicant providing the results of the detailed control room design review for Unit 1.
June 7, 1982	Letter from applicant responding to SER Confirmatory Issue (22).
June 8, 1982	Letter from applicant responding to SER Confirmatory Issues (23) and (24).
June 9, 1982	NRC letter identifying concerns regarding the use of CO_2 in the control room.
June 9, 1982	Letter from applicant summarizing the reactor vessel structural analysis for control of heavy loads.
June 14, 1982	NRC letter identifying Mark III containment design safety issues.
June 16, 1982	Letter from applicant responding to SER Outstanding Issue (15).
June 22, 1982	Letter from applicant describing design modifications for "fast scram" hydrodynamic loads on control rod drive (CRD) systems.
July 20, 1982	Letter from applicant regarding SER Outstanding Issue (12).
July 21, 1982	Letter from applicant requesting extension of Units 1 and 2 construction completion schedules.
July 28, 1982	NRC letter regarding unresolved fire protection safety issues in the SER.
July 30, 1982	NRC letter providing agenda for Unit 1 in-progress control room audit.
July 30, 1982	NRC letter providing the Advisory Committee on Reactor Safeguards (ACRS) report on Unit 1.
August 5, 1982	Letter from applicant regarding Mark III containment design safety issues.
August 5, 1982	Letter from applicant revising operating license applications to 40 years from date of issuance of the licenses.
August 12, 1982	NRC letter regarding qualification of safety-related equipment for hydrodynamic loads.
August 13, 1982	Letter from applicant summarizing standby liquid control and other design features for mitigation of anticipated transients without scram.

August 16, 1982	Letter from applicant regarding feedwater check valve function following a line break outside containment.
August 16, 1982	Letter from applicant providing containment annulus concrete design, construction, and testing.
August 18, 1982	Letter from applicant providing additional information regarding "fast scram" hydrodynamic loads on CRD systems.
August 18, 1982	NRC letter issuing SER Supplement No. 1.
August 18, 1982	Letter from applicant regarding the environmental qualification program.
August 20, 1982	Letter from applicant responding to SER Confirmatory Issue (15).
August 25, 1982	Letter from applicant submitting Amendment 8 to the FSAR.
August 26, 1982	Letter from applicant responding to SER Confirmatory Issue (34).
August 26, 1982	Letter from applicant responding to ACRS report on Unit 1.
August 30, 1982	NRC letter requesting additional information on the Emergency Plan.
August 30, 1982	Letter from applicant providing additional information concerning qualification of safety-related equipment to hydrodynamic loads.
August 31, 1982	Letter from applicant responding to SER Outstanding Issue (16).
August 31, 1982	Letter from applicant responding to SER Confirmatory Issues (36), (37), (38), and (39).
September 1, 1982	Letter from applicant responding to SER Outstanding Issues (13) and (14).
September 2, 1982	Letter from applicant submitting the preservice inspection program plan.
September 8, 1982	NRC letter identifying SER Outstanding Issue (21).
September 9, 1982	Letter from applicant regarding SER Outstanding Issue (3).
September 9, 1982	Letter from applicant regarding SER Outstanding Issue (8).
September 9, 1982	Letter from applicant documenting position on fuel issues - SER Confirmatory Issues (11), (12), (13), and (14).

September 9, 1982	Letter from applicant providing additional information on emergency planning.
September 13, 1982	Letter from applicant regarding SER Confirmatory Issues (46), (47), (48), and (49).
September 15, 1982	Letter from applicant responding to Idaho Nuclear Engineering Laboratory draft technical evaluation report on control of heavy loads (NUREG-0612).
September 15, 1982	Letter from applicant providing advance copies of FSAR Sections 3.10 and 3.11.
September 16, 1982	NRC letter requesting additional information regarding degraded core hydrogen control.
September 16, 1982	Letter from applicant committing to several Licensing Review Group-II generic issue positions.
September 17, 1982	Letter from applicant providing additional information on seismic and dynamic qualification of mechanical and electrical equipment.
September 20, 1982	Letter from applicant providing Revision 2 to the Fire Protection Evaluation Report.
September 22, 1982	Letter from applicant providing the revised Perry corporate quality assurance (QA) program.
September 22, 1982	Letter from applicant providing Revision 0 of the Emergency Plan.
September 22, 1982	Letter from applicant responding to SER Confirmatory Issue (16).
September 28, 1982	Letter from applicant responding to SER Outstanding Issue (21).
September 29, 1982	Letter from applicant providing additional information regarding degraded core hydrogen control.
September 30, 1982	Letter from applicant submitting Amendment 9 to the FSAR.
October 5, 1982	NRC letter clarifying request for additional information in the resolution of SER Outstanding Issue (21).
October 8, 1982	Letter from applicant providing reactor internals vibration prototype test program.
October 8, 1982	Letter from applicant providing Kuosheng safety/relief valve (SRV) test data.

October 14, 1982	Letter from applicant responding to SER Confirmatory Issues (28),(30), and (33).
October 14, 1982	Letter from applicant responding to SER Confirmatory Issues (32) and (45).
October 15, 1982	Letter from applicant providing additional information regarding SRV hydrodynamic loads.
October 19, 1982	NRC letter requesting clarification and additional information regarding QA lists documented in FSAR Amendments 8 and 9.
October 25, 1982	Letter from applicant responding to SER Confirmatory Issue (27).
November 3, 1982	NRC letter transmittng report of October 6, 1982 emergency preparedness meeting.
November 4, 1982	NRC letter requesting additional information to resolve SER Outstanding Issue (12).
November 8, 1982	Letter from applicant amending September 15, 1982 letter regarding control of heavy loads (NUREG-0612).
November 8, 1982	Letter from applicant regarding containment annulus concrete design requirements.
November 12, 1982	NRC letter providing the staff's Unit 1 control room design interim audit report.
November 16, 1982	Letter from applicant providing additional information regarding SER Confirmatory Issue (36).
November 16, 1982	Letter from applicant regarding SER Outstanding Issue (21).
November 17, 1982	Letter from applicant providing additional information on SRV hydrodynamic loads.
November 23, 1982	Letter from applicant submitting Amendment 10 to the FSAR.
November 26, 1982	NRC letter on staff's findings regarding SER Outstanding Issues (15) and (16) and Confirmatory Issue (36).
November 30, 1982	NRC letter requesting automatic depressurization system modifications commitment to TMI Action Plan Item II.K.3.18 (SER Confirmatory Issue (25)).
November 30, 1982	NRC letter requesting information for equipment qualification audit scheduling.

December 1, 1982	Letter from applicant requesting extension of construction permit completion dates for Units 1 and 2.
December 2, 1982	Letter from applicant responding to SER Confirmatory Issue (20) and Outstanding Issue (10).
December 2, 1982	Letter from applicant identifying management changes effective December 1, 1982.
December 14, 1982	NRC letter requesting additional information regarding SRV piping and quencher device.
December 21, 1982	Letter from applicant regarding SER Outstanding Issue (21).
December 21, 1982	Letter from applicant responding to SER Confirmatory Issues (41) and (42).
December 21, 1982	Letter from applicant regarding SER Confirmatory Issue (44).
December 21, 1982	Letter from applicant responding to QA question posed in NRC letter dated October 19, 1982.
December 23, 1982	NRC letter regarding proposed changes to the Perry Environmental Monitoring Program.
December 27, 1982	NRC letter transmitting draft Standard Technical Specification for BWR/6 plants.
December 29, 1982	NRC Order extending construction completion dates for Perry, Units 1 and 2.

APPENDIX B

REFERENCES*

American Industrial Hygienists Association, Hygienic Guide, 1964.

Bush, S. H., "Probability of Damage to Nuclear Components," Nuclear Safety, 14(3):187, May-June 1973.

---, "A Reassessment of Turbine-Generator Failure Probability," Nuclear Safety, 19(6):681, Nov.-Dec. 1978.

Cleveland Electric Illuminating Company, "Perry Nuclear Power Plant, Unit Nos. 1 and 2, Final Safety Analysis Report," Docket Nos. 50-440/441, Jan. 30, 1981

Code of Federal Regulations, Title 10 (10 CFR), "Energy" (includes General Design Criteria).

General Electric Topical Report NEDE-22203-P, "Reactor Internals Vibration Predictions," Aug. 1982. (Proprietary. Not publicly available.)

---, NEDO 10466, "Power Generation Control Complex Design Criteria and Safety Evaluation," Rev. 2, Mar. 1978.

Letter, Aug. 9, 1976 from G. G. Sherwood (General Electric) to B. C. Rusche (NRC), Subject: Selection of BWR/6 Relief Valve Control System Modifications.

---, Dec. 22, 1980 from D. G. Eisenhut (NRC) to operating license holders, Subject: Submission of NUREG-0612.

---, May 17, 1982 from D. L. Holtzscher (Illinois Power Company) to H. J. Faulkner (NRC), Subject: Submittal of Position Papers for Licensing Review Group-II Issues.

Memorandum, Aug. 19, 1982 from L. S. Rubenstein to T. M. Novak, Subject: Resolution of LRG-II Channel Box Deflection Issue (LRG-II Issue 3-CPB).

National Institute of Occupational Safety and Health, "Occupation Health Guide for Carbon Dioxide," Publication 81-123, Washington, DC, 1981.

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

*All correspondence between the applicant and the NRC referenced in this supplement is listed in Appendix A of this supplement.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," Nov. 1980.

---, NUREG-0800 (formerly NUREG-75/087), "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants," 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 and Supplement No. 1, Aug. 1982.

---, Regulatory Guide 1.8, "Personnel Selection and Training," Rev. 1, May 1977.

---, Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment," Feb. 1982.

---, Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," Rev. 2, May 1976.

---, Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," July 1977.

---, Regulatory Guide 1.117, "Tornado Design Classification," June 1976.

U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Bulletin 79-02, "Pipes Support Base Plate Designs Using Concrete Expansion Anchor Bolts," Mar. 2, 1979.

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

---, Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Controls," Mar. 13, 1980.

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American Institute of Steel Construction (AISC), "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," Sixth Edition, New York, N.Y., 1969.

American National Standards Institute (ANSI), N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel," 18.1-1978 (p. 13-14).

American Society for Testing and Materials (ASTM), E-185-73," Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels."

American Society of Mechanical Engineers (ASME), "Boiler and Pressure Vessel Code," Section III, "Nuclear Power Plant Components," Appendix G, "Protection Against Non-Ductile Failure."

---, Section III, Subsection NF.

---, Section XI, "Rules for Inservice Inspection of Nuclear Reactor Coolant Systems."

Institute of Electrical and Electronics Engineers, Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

APPENDIX C

UNRESOLVED SAFETY ISSUES

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

Task A-11 Reactor Vessel Materials Toughness

In Section C.5, Appendix C, of the SER, the staff identified, as an unresolved safety issue (USI), reactor vessel materials toughness. On the basis of its evaluation of the Perry reactor vessel materials, the staff concluded that Perry could operate before resolution of this generic issue without undue risk to the health and safety of the public. Since issuance of the SER and Supplement No. 1 to the SER, the staff has completed its generic assessment of this USI. The results of this effort have been documented in NUREG-0744, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," Volumes I and II, Revision 1, published in October 1982.

NUREG-0744 provides the staff's position with respect to the reactor vessel safety analysis according to the rules given in the Code of Federal Regulations, Title 10, which requires that an analysis be performed whenever neutron irradiation reduces the Charpy V-notch upper-shelf energy level in the vessel steel to 50 ft-lb or less. Task A-11 was undertaken because the available engineering methodology for such an analysis used linear elastic fracture mechanics principles, which could not fully account for the plastic deformation or stable crack extension expected at upper-shelf temperatures. The goal of Task A-11 was to develop an elastic-plastic fracture mechanics methodology applicable to the beltline region of a pressurized-water-reactor vessel that could be used in the required safety analysis. This goal was achieved with the help of a team of recognized experts.

Therefore, NUREG-0744, Revision 1, completes the staff's resolution of USI A-11. The information contained therein will be the basis for licensing actions taken by the NRC relative to the toughness requirements set forth in 10 CFR 50, Appendix G. NRC generic letter (No. 82-26), dated November 12, 1982, encourages CEI to review NUREG-0744, Revision 1, and consider its application in those cases where it may be necessary to submit a fracture analysis for staff review and approval.

C.6 References

Code of Federal Regulations, Title 10 (10 CFR), "Energy."

Letter, Nov. 12, 1982, from D. G. Eisenhut (NRC) to all power reactor licensees, Subject: NUREG-0744, Rev. 1 - Pressure Vessel Material Fracture Toughness (Generic Letter 82-26).

U.S. Nuclear Regulatory Commission, NUREG-0744, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," Vols. I and II, Rev. 1, Oct. 1982.

APPENDIX E

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This supplement to the Safety Evaluation Report is a product of the NRC staff. NRC staff members listed below were principal contributors to this supplement.

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APPENDIX G

ERRATA TO THE SAFETY EVALUATION REPORT

Page	Section/Table	Change
1-11	1.11	For Item (4) delete asterisks.
1-13/1-14	Table 1.1	Under column, "SER Section," for "Issue 4-RSB," add "6.3.1.3"; "Issue 6-RSB," change "5.4.7" to "5.4.2"; "Issue 1-CPB," change "4.2.3.2" to "4.2.3.3"; "Issue 2-CPB," change "4.2.1.3" to "4.2.3.2" and "4.2.4.3" to "4.2.3.3"; "Issue 2-CSB," change "6.2.5" to "6.2.7"; "Issue 4-ASB," change "9.4.5.4" to "9.4.5.3." Under column, "Title (TMI Action Plan Item)" for "Issue 8-RSB," add "(II.K.3.28)" at end of the title.
1-17	Table 1.2	Add new item, "II.K.3.28; Qualification of ADS Accumulators; 6.3.1.3" under the respective columns.
3-37	3.9.3.1	At the end of the last sentence of paragraph 4, add the following: "when the applicant responds to Outstanding Issue (6) - see Section 6.2.1.8 of this report".
5-30	5.4.2	In the third full paragraph, last line, change "6.3.2.3" to "6.3.1.3".
6-17	6.2.7(3)	In Item (f), change "6.2.4" to "3.9.3.2.1", and at the end of the sentence add "which will be addressed separately in the resolution of Confirmatory Issue (7)."
6-22	6.3.1.3	In third full paragraph, last line, after "check valves," add "to meet the requirements of TMI Action Plan Item II.K.3.28 (also LRG-II Generic Issue 8-RSB)".
6-24	6.3.1.3	In paragraph 4, line 5, change "8-RSB" to "4-RSB".

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16. ABSTRACT *(200 words or less)*

Supplement No 2 to the Safety Evaluation Report 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, 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 facility is located near Lake Erie in Lake County, Ohio. This supplement has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission and reports the status of certain items that had not been resolved at the time of publication of the Safety Evaluation Report.

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