NUREG-0853
Supplement No. 4

Safety Evaluation Report
related to the operation of Clinton Power Station, Unit No. 1

Docket No. 50-461

Illinois Power Company, et al.

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

February 1985

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ABSTRACT

Supplement No. 4 to the Safety Evaluation Report on the application filed by Illinois Power Company, Soyland Power Cooperative, Inc., and Western Illinois Power Cooperative, Inc., as applicants and owners, for a license to operate the Clinton Power Station, Unit No. 1, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Harp Township, DeWitt County, Illinois. This supplement reports the status of items that have been resolved by the staff since Supplement No. 3 was issued.

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1 INTRODUCTION AND GENERAL DESCRIPTION

1.1 Introduction

The Nuclear Regulatory Commission staff (referred to as the NRC staff or staff) issued its Safety Evaluation Report (SER) (NUREG-0853) in February 1982 regarding the application by Illinois Power Company et al. (hereinafter referred to as the applicant) for a license to operate the Clinton Power Station, Unit 1, Docket No. 50-461. Supplement No. 1 (SSER 1) to the Clinton SER was issued in July 1982; SSER 2 was issued in May 1983; and SSER 3 was issued in May 1984. The purpose of this supplement, No. 4 (SSER 4), is to further update the SER by providing results of the NRC staff's review of information submitted by the applicant to address some of the unresolved issues listed in Sections 1.9 and 1.10 of the SER.

Each section and appendix of this supplement is numbered and titled so that it corresponds to the section or appendix of the SER that is relevant to the NRC staff's additional evaluation. Except where specifically noted, the material in this supplement does not replace the material in corresponding SER section or appendix. Appendix A is a continuation of the chronology of correspondence between NRC and the applicant and updates the lists in the SER and SSER 1 through SSER 3. Appendix B is a list of references cited in this report.* Appendix D is a list of principal staff contributors to this supplement.

Copies of this SER supplement are available for inspection at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C., and at the Warner Vespasian Library, Clinton, Illinois. Copies are also available for purchase from the sources indicated on the inside front cover.

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1.9 Outstanding Issues

In SER Section 1.9, the NRC staff identified twenty outstanding issues that had not been resolved at the time the document was issued. SSER 1 reported that four of those items had been satisfactorily resolved and one had been changed

*The availability of the material cited is described on the inside front cover of this report.

to a confirmatory status.* SSER 2 reported that six items had either been resolved or changed to a confirmatory status. SSER 3 reported that four items had been resolved. Therefore, six outstanding issues remained that had not yet been resolved after the issuance of SSER 3.

In SSER 3 the status of outstanding issue 13, "Remote shutdown system," was stated as resolved. This issue was identified as requiring resolution from two aspects in the SER: (1) Compliance with GDC 19 (SER Section 7.4.3.1) as it relates to redundant safety-grade capability to achieve and maintain hot shutdown from a location remote from the control room and (2) the provision of a safe shutdown analysis for fire protection that satisfies the fire protection technical requirements for safe shutdown contained in Sections III.G and III.L of Appendix R to 10 CFR 50 (SER Section 9.5.5). In SSER 3, only the fire protection aspect of this issue was addressed; therefore, this issue has been reopened to resolve the compliance with the GDC 19 aspect. Since this issue is being reopened, seven outstanding issues remain.

The present supplement (SSER 4) partially resolves one outstanding issue. The current status of each of the twenty original issues is tabulated below. For those items discussed in this supplement, the relevant sections in this document are indicated. Resolution of issues that are, to date, unresolved will be reported in future supplements.

Issue		Status	SSER 4 Section(s)
(1)	Transportation accidents	Resolved in SSER 3	--
(2)	Effects of Unit 2 excavation	Resolved in SSER 2	--
(3)	Seismic analysis	Became confirmatory issue 70, resolved in SSER 3	--
(4)	Internally generated missiles	Resolved in SSER 1	--
(5)	Postulated piping failures	Under review	--
(6)	Steady-state vibration acceptance criteria for balance of plant piping	Resolved in SSER 2	--
(7a)	Environmental qualification of electrical and mechanical equipment	Under review	--
(7b)	Seismic and dynamic qualification of mechanical and electrical equipment	Under review	--

*SSER 2 stated that only three items had been closed and it was silent regarding confirmatory status.

Issue		Status	SSER 4 Section(s)
(7c)	Pump and valve operability qualification	Under review	--
(8)	Preservice (PSI) and inservice inspection (ISI) programs	PSI program: became confirmatory issue 67 in SSER 1 ISI program: became license condition 12 in SSER 2	--
(8a)	Preservice and inservice testing of pumps and valves	Became confirmatory issue 68 in SSER 1	--
(9)	Pool dynamic loads	Under review	--
(10a)	Containment purge	Became confirmatory issue 69 in SSER 2	--
(10b)	Containment isolation	Resolved in SSER 2	--
(10c)	Containment leakage testing (vent and drain lines)	Resolved in SSER 2	--
(10d)	Containment leakage testing (secondary containment)	Resolved in SSER 2	--
(10e)	Containment bypass leakage	Resolved in SSER 2	--
(11)	Control room habitability	Resolved in SSER 1	--
(12)	Engineered safety features reset controls (IE Bulletin 80-06)	Resolved in SSER 2	--
(13)	Remote shutdown system	Partially resolved in SSER 3	--
(14)	Capability for safe shutdown following loss of bus supplying power to instruments and controls (IE Bulletin 79-27)	Resolved in SSER 2	--
(15)	Control system failures resulting from high-energy-line breaks or common power source or sensor malfunctions	Under review	--
(16)	Separation of the RPS and MSIV solenoid circuits and PGCC circuits	Resolved in SSER 1	--

Issue	Status	SSER 4 Section(s)
(17) Organization and staffing	Under review	--
(18a) Onsite emergency plan	Resolved in this SSER	13.3
(18b) Offsite emergency plan	Awaiting information	--
(19) Security	Resolved in SSER 1	--
(20) QA program	Resolved in SSER 3	--

1.10 Confirmatory Issues

In SER Section 1.10, the NRC staff identified 66 confirmatory issues for which additional information and documentation were required to confirm preliminary conclusions. SSER 1 reported that 28 of those items had been satisfactorily resolved. SSER 2 addressed 11 additional issues that have been resolved, as well as certain issues that still require resolution. SSER 3 addressed 9 additional issues that have been resolved. The present supplement (SSER 4) partially resolves two and totally resolves ten confirmatory issues. The current status of each of the 66 original issues is tabulated below. Four issues (67, 68, 69, and 70) that previously had been outstanding issues in SSER 1 were added to the confirmatory list in SSER 2. Resolution of confirmatory issues that are, to date, unresolved will be reported in future supplements.

Issue	Status	SSER 4 Section(s)
(1) Emergency preparedness meteorological program	Under review - Section 2.3.3 updated	--
(2) Inspection program around the ultimate heat sink (UHS) and the main cooling lake dam	Resolved in SSER 1	--
(3) Protection of UHS dam abutments against soil erosion	Resolved in SSER 1	--
(4) Internally generated missiles - fan failures	Resolved in SSER 2	
(5) Design adequacy of cable tray system	Resolved in SSER 1	--
(6) Containment ultimate strength analysis	Removed from list in this SSER	3.8.1
(7) Structural integrity of safety-related masonry walls	Resolved in SSER 2	--
(8) NSSS pipe break analysis using SRP criteria	Resolved in SSER 1	--

Issue	Status	SSER 4 Section(s)
(9) Vibration assessment of RPV internals	Resolved in this SSER	3.9.2
(10) Annulus pressurization loads (LOCA asymmetric loads)	Resolved in this SSER	3.9.2
(11) Use of SRSS for combining Mark III dynamic responses for other than LOCA and SSE	Resolved in SSER 1	--
(12) IE Bulletin 79-02 regarding support baseplate flexibility	Resolved in SSER 2	--
(13) Mark III hydrodynamic loads	Became part of outstanding issue 9 to avoid duplication in this SSER	3.9.3.1
(14) Feedwater check valve analysis	Resolved in SSER 2	--
(15) Seismic and LOCA loadings on fuel assemblies (LRG II Issue 2-CPB)	Resolved in this SSER	4.2.3.4
(16) Scram discharge system evaluation	Resolved in SSER 1	--
(17) Fracture toughness data	Resolved in SSER 1	--
(18) Subcompartment pressure analysis	Under review	--
(19) Combustible gas control	Resolved in SSER 3	--
(20) Containment isolation dependability	Resolved in SSER 2	--
(21) Containment monitoring, II.F.1(1) through II.F.1(6)	Partially resolved in this SSER (II.F.1(1) and II.F.1(2) only remaining issues)	6.2.7, 12.3.4.1
(22) Plant-specific LOCA analysis, II.K.3.31	Resolved in SSER 3	--
(23) High drywell pressure interlocks	Resolved in SSER 1	--
(24) ATWS recirculation pump trip	Awaiting information	--
(25) Response-time testing	Resolved in SSER 1	--

Issue	Status	SSER 4 Section(s)
(26) Analog trip modules and optical isolators	Resolved in SSER 2	--
(27) Susceptibility of the NSPS to electrical noise	Resolved in SSER 1	--
(28) Modification of ADS logic, II.K.3.18	Resolved in this SSER	7.3.3.4
(29) Restart of low-pressure systems, II.K.3.21	Resolved in SSER 1	--
(30) Temperature effects on level measurements	Resolved in SSER 2	--
(31) Containment atmosphere monitoring system	Resolved in this SSER	7.6.3.1
(32) Verification that testing is in accordance with BTP PSB-1	Removed from list in SSER 1	--
(33) Electrical drawing review	Removed from list in SSER 1	--
(34) Verification of diesel generator testing	Resolved in this SSER	8.3.1
(35) Class A supervision and power supply for fire detection system	Resolved in SSER 3	--
(36) Circulating water system	Resolved in SSER 2	--
(37) Initial test program	Resolved in SER	--
(38) Human engineering aspects of control room design, I.D.1	Under review	--
(39) Common reference for reactor vessel level instruments, II.K.3.27	Resolved in SSER 2	--
(40) Shielding design review, II.B.2	Resolved in SSER 1	--
(41) Short-term accident and procedures review, I.C.1, I.C.7, I.C.8	Partially resolved in this SSER (I.C.1 only remaining issue)	13.6.3
(42) Training during low-power testing, I.G.1	Awaiting information	--

Issue	Status	SSER 4 Section(s)
(43) Review ESF values, II.K.1.5	Resolved in SSER 1	--
(44) Operability status, II.K.1.10	Resolved in SSER 1	--
(45) HPCI and RCIC initiation levels, II.K.3.13	Resolved in this SSER	6.3.2.3
(46) Isolation of HPCI and RCIC, II.K.3.15	Resolved in this SSER	7.3.3.3
(47) Qualification of ADS accumulators, II.K.3.28	Under review	--
(48) Plant-specific analysis, II.K.3.30	Resolved in SSER 3	--
(49) ODYN analysis for River Bend as applied to Clinton	Resolved in SSER 1	--
(50) Conformance evaluation report for loose-parts monitoring system	Resolved in SSER 3	--
(51) Requirements of NUREG-0313	Resolved in SSER 1	--
(52) Control room habitability - chlorine gas	Resolved in SSER 1	--
(53) Debris screen design	Resolved in SSER 2	--
(54) Verification of adequacy of fire protection systems	Removed from list in SSER 1	--
(55) Flood-proof door	Resolved in SSER 2	--
(56) Valves in fire protection water supply system	Resolved in SSER 1	--
(57) Break in water supply piping	Resolved in SSER 1	--
(58) Test data on fire ratings	Resolved in SSER 3	--
(59) Three-hour-fire-rated penetration seals	Resolved in SSER 3	--
(60) Install fire protection equipment (emergency lighting)	Resolved in SSER 3	--
(61) Fire protection administrative controls and training	Resolved in SSER 1	--

Issue	Status	SSER 4 Section(s)
(62) Technical Specification on fire protection	Resolved in SSER 1	--
(63) Periodic leak testing of pressure isolation values	Resolved in SSER 1	--
(64) Sedimentation in UHS	Resolved in SSER 1	--
(65) Protection against postulated piping failures	Resolved in SSER 1	--
(66) Steam bypass of the suppression pool (LRG II Issue 3-CSB)	Under review	--
(67) Preservice inspection program	Under review	--
(68) Preservice testing of pumps and valves	Under review	--
(69) Containment purge	Under review	--
(70) Seismic analysis	Resolved in SSER 3	--
(71) Humphrey concerns	Under review	--

1.11 License Conditions

In SER Section 1.11, the NRC staff identified nine potential license conditions that may be required as part of the operating license for Clinton Unit 1 to ensure that NRC requirements are met during plant operations. Two additional potential license conditions (10 and 11) were identified in SSER 1, and SSER 2 identified two additional conditions (12 and 13), as well as one (6) for which additional requirements were imposed. One condition (14) was added in SSER 3. The current status of these issues and the sections in which they are resolved are shown below.

Issue	Status	SSER 4 Section(s)
(1) Staffing DeWitt pumping station	Under review	--
(2) New stability analysis before second cycle of operation	Awaiting information	--
(3) Postaccident monitoring	Under review	--
(4) Vacuum relief valve position indication	Awaiting information	--
(5) Hydrogen management	Under review	--

Issue	Status	SSER 4 Section(s)
(6) Postaccident sampling, II.B.3	Under review	--
(7) Diesel generator reliability	Awaiting information	--
(8) Kuosheng-1 test program	Resolved in this SSER	3.9.2, 6.2.1.8
(9) Visual examination of discharged fuel	Under review	--
(10) Measurement of groundwater level	Under review	--
(11) Security	Under review	--
(12) Inservice inspection	Under review	--
(13) Control of heavy loads	Under review	--
(14) Transportation accidents	Awaiting information	--

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

In SSER 1 the staff requested that the applicant confirm that the pressure-resisting capacity of the seals around the airlock doors and equipment hatches was 69 psig. By letter dated March 15, 1983, the applicant provided the confirmation requested by the staff that the type of closures used on the Clinton Unit 1 containment has been tested at 69 psig with no apparent leakage.

In SSER 1 it was indicated that the applicant's position related to the effects of local detonation on the containment structure and its penetrations is under review by the staff. Since then the staff has determined that the likelihood of local detonations is so remote that structural response analyses are not needed at this time. Therefore, this issue will be reviewed under TMI Action Plan Item II.B.8 and removed from the list of confirmatory issues.

In addition to ultimate pressure capacity for the containment/drywell structures for positive pressure, the applicant provided in a September 27, 1984, letter, in response to the staff's request that was made in a letter dated June 18, 1984, the ultimate pressure capacities for negative pressure. For the containment, the capacity for negative pressure as given by the applicant is -11 psig and the steel liner is the governing element. The ultimate capacity for the drywell pressure-retaining boundary for negative pressure as determined by the applicant is -61 psid and the governing element is the drywell personnel airlock door's main hinge pin. The staff has reviewed the criteria used in establishing the ultimate capacities for the containment and drywell, and found them to be conservative. The ultimate capacities for negative pressures as determined by the applicant are, therefore, acceptable.

3.9 Mechanical Systems and Components

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

In Section 3.9.2 of the SER for Clinton (NUREG-0853) the applicant was required to provide a summary of the results of the Kuosheng Nuclear Power Plant Unit 1 (Kuosheng Unit 1) reactor internals vibration test data for the staff review. This was identified as confirmatory issue 9 and part of licensing condition 8 in Section 1.10 of the SER. The applicant has provided the required data in the form of Report No. NEDE-22146, "Kuosheng-1 Reactor Internals Vibration Measurements," dated July 1982 (enclosure to Sept. 16, 1983, letter from applicant). This report presents the results of the vibration measurements which were made on reactor internal components for the General Electric prototype boiling water reactor, BWR/6-218, at Kuosheng Unit 1 in Taiwan from September 26, 1980, through November 11, 1981. The NRC staff reviewed this report and concluded that Kuosheng Unit 1 can be used as a valid prototype for the Clinton plant for the following reasons:

In Section 3.9.2.4 of the FSAR, the applicant stated that the reactor internals for Clinton Unit 1 are of the same design as Kuosheng Unit 1. The applicant has also reported, in a letter dated September 16, 1983, and in the FSAR, that vibration and inspection measurements were conducted for the prototype 218-inch size BWR/6 reactor at the Kuosheng Unit 1 plant in accordance with the guidelines of Regulatory Guide (RG) 1.20. These tests were conducted in three phases: preoperational tests prior to fuel loading, zero-power tests with fuel, and initial startup tests, as described in the SER Section 3.9.2. Vibration sensors included strain gauges, displacement sensors (linear variable transformers), and accelerometers. As reported by the applicant in the FSAR, comparisons of measured vibration amplitudes with predicted and allowable amplitudes showed that all vibrations were within the established criteria (letter from applicant, Sept. 16, 1983). Since the reactor internals for Clinton Unit 1 are reported to be of the same design as those of Kuosheng Unit 1 and the latter have been tested and found to satisfy the regulatory requirements as stated above, Kuosheng Unit 1 can be accepted as a valid prototype for Clinton Unit 1.

The applicant has committed to inspect the Clinton reactor internals in accordance with the requirements of RG 1.20, Rev. 2, Paragraph 3.1.3, for nonprototypes. Preoperational flow tests will also be conducted at the same steady-state conditions and for the same duration as at the Kuosheng plant.

After the Clinton SER was issued, the NRC staff reviewed and approved the LRG-II position paper regarding modifications of BWR/6 internals that were intended to prevent fatigue failure (breakage) of incore instrument tubes from flow-induced vibration. The fatigue failure problem was identified in the Kuosheng BWR/6-218 reactor following an inadvertently sustained operation of the RHR/LPCI system for an extended period of time. The modifications consisted of (1) installation of flow deflector plates at the LPCI inlets to the core shroud and (2) replacement of the intermediate range monitoring (IRM) tubes in locations near the LPCI injection inlets with strengthened tubes of improved design.

While approving the proposed LPCI modifications, the staff requested that the applicant confirm that the LPCI modifications and certain operational controls would be implemented for the Clinton BWR/6 facility. These operational controls were concerned with the instructions to be given to plant operators not to operate the RHR system in the LPCI mode unless it was required for an accident, emergency, or for short-term testing situations. These operational controls also required the applicant to report to the NRC the circumstances of any inadvertent operation of the modified RHR/LPCI system for an extended period of time. The applicant has confirmed (letter, Sept. 30, 1983) that the RHR/LPCI modifications and the related operational controls will be implemented at the Clinton Unit 1 plant.

On the basis of the staff's acceptance of the Kuosheng Unit 1 plant as the valid prototype for Clinton Unit 1 and documented confirmation that the RHR/LPCI modifications and related operational controls will be implemented at Clinton Unit 1, confirmatory issue 9 is considered to be resolved.

In addition, since the Kuosheng Unit 1 reactor internals test is a valid prototype for Clinton Unit 1 and in Section 6.2.1.8 of this SSER the staff determined that the applicability of the Kuosheng SRV quencher design to Clinton Unit 1 has been demonstrated, licensing condition 8 is considered to be resolved.

In Section 3.9.2 of the SER, the staff reported that the applicant had not yet provided information to demonstrate that certain reactor system components and their supports can withstand the dynamic effects of postulated asymmetric LOCA loads. This was subsequently identified as confirmatory issue 10 in Section 1.10 of Supplement No. 1 to the SER. In response to requests from the staff, the applicant provided this information in Amendments 15 and 20 to the FSAR.

The response to asymmetric LOCA loads (annulus pressurization loads) was calculated for reactor coolant system piping and components, including their supports, and for reactor internals. These responses were included in appropriate loading combinations and the resulting stresses were all below the applicable ASME Code Service Limits. Therefore, the staff concludes that confirmatory issue 10 has been resolved.

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

3.9.3 1 Loading Combinations, Design Transients, and Stress Limits

In the SER the staff stated that the Mark III hydrodynamic loads are yet to be reviewed and determined as part of unresolved safety issues A-39, "Determinatior of Safety/Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containment," and B-10, "Behavior of BWR Mark III Containment." Since this issue is duplicated in Section 6.2.1.8.2 of the SER, it will no longer be addressed in this section.

4 REACTOR

4.2 Fuel System Design

4.2.3 Design Evaluation

4.2.3.4 Seismic and LOCA Loadings

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 and Appendix A to that section state that fuel system coolability should be maintained and that damage (including liftoff) should not be so severe as to prevent control rod insertion when it is required during these low probability accidents. General Electric Co. (GE) has described the entire seismic and LOCA loadings evaluation in Topical Report NEDE-21175-3.

In the SER the staff stated the applicant must confirm that plant-specific seismic and LOCA loadings on the fuel assemblies are bounded by the loadings used in GE Topical Report NEDE-21175-3, and provide a liftoff analysis or other confirmation that the fuel assembly liftoff under externally applied loads will not be large enough to alter the horizontal alignment supplied by the lower tie plates.

In a letter from C. O. Thomas (NRC) to J. F. Quirk (GE), dated October 20, 1984, the staff generically approved NEDE-21175-3 for combined seismic and LOCA loading analysis. In addition, the staff has completed its review of the applicant's plant-specific values of liftoff and accelerations described in the FSAR Amendment 29 submittal. The results of the applicant's analysis show that the vertical liftoff occurs within the allowable limit specified in NEDE-21175-3 and the vertical and horizontal accelerations are within the evaluation-basis limits described in NEDE-21175-3, thereby assuring structural integrity and control rod insertability during seismic and LOCA events.

The applicant has provided the information necessary to resolve the confirmatory issue related to seismic and LOCA loadings on the reactor fuel assemblies for Clinton Unit 1.

4.4 Thermal and Hydraulic Design

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

In the SER the staff stated that the instrumentation for detecting inadequate core cooling (including existing level instrumentation and incore thermocouples) are either under staff review or will be reviewed upon receipt of the applicant's submittal.

The BWROG has submitted two reports: SLI-8211 (S. Levy Inc.), dated July 1982, and SLI-8218 (S. Levy Inc.), dated December 1982; the applicant (in letters dated Oct. 10, 1984, and Dec. 5, 1984), has provided responses addressing three water level instrumentation concerns identified in SLI-8211.

The present NRC position on the issue of the detection of inadequate core cooling (ICC) is that if the applicant upgrades the water level system to be consistent with the recommendations in SLI-8211, then there is no additional instrumentation required for ICC detection. Previously, as specified in an earlier revision of Regulatory Guide (RG) 1.97, in-core thermocouples were to be installed in boiling-water reactors (BWRs). However, in 1981 the ACRS recommended (as stated on page 2 of ACRS Report No. 0938, dated August 11, 1981) that installation of in-core thermocouples be reevaluated. The BWROG also submitted information (contained in Appendix B of SLI-8218 which was docketed at a later date) that concluded the effectiveness of in-core thermocouples as an ICC indicator is very limited, and led the BWROG to recommend to the staff that in-core thermocouples not be used to detect ICC.

The staff, in reviewing the BWROG recommendation, questioned the reliability of existing water level instrumentation as the sole indication of ICC, and requested that the BWROG perform a further study to evaluate the need for upgrading existing water level instrumentation to make it more reliable as an ICC detector. The staff also requested that the BWROG consider what other instrumentation (including in-core thermocouples) might be needed in the BWR plant monitoring system. To reflect the review status, Revision 3 of RG 1.97, dated May 1983, deleted the provision for installation of in-core thermocouples. Instead of installation of in-core thermocouples, the staff provided BWR applicants an opportunity to demonstrate that other available means of detecting ICC are adequate.

In response, report SLI-8211 (which includes the BWROG's evaluation of existing water level instruments and recommendations for their improvement) and report SLI-8218 (which includes the results of an evaluation of additional instrumentation as diverse indicators of ICC and recommendations regarding the need for such additional instrumentation) were submitted for staff review. In addition, at the staff's request, the applicant also submitted a plant-specific evaluation (in letters dated Oct. 10, 1984, and Dec. 5, 1984) addressing the applicability of BWROG's findings (in reports SLI-8211 and SLI-8218) to Clinton.

The staff has completed its review of BWROG report SLI-8211; the results of that review are included in the NRC Generic Letter 84-23 dated October 26, 1984. The staff also has reviewed the applicant's response (in letters dated Oct. 10, 1984, and Dec. 5, 1984) describing modifications to the water level measurement system to make it more reliable during postulated accident conditions. The modifications included re-routing of instrument sensing lines within the drywell to limit the overall vertical drop to within 30 inches and relocation of the instrument line flow limiting orifice plates to near the corresponding drywell penetration. The applicant also stated that Clinton already uses analog trip units rather than less reliable mechanical types, and that the Clinton logic design (for reactor trip and/or engineered safety feature (ESF) systems(s) actuation on reactor vessel low water level) has four divisions and is identical to Plant B in SLI-8211. In the SLI-8211 review of Plant B, there were no cases identified which failed to provide automatic reactor trip and ECCS actuation; therefore, the staff concludes that no changes are required for the Clinton protection system logic. On the basis of the review, the staff concludes that the Clinton water level measurement system is in compliance with the BWROG's recommendations in SLI-8211 and is, therefore, acceptable.

The staff has also completed its review of the SLI-8218 and agrees that the application of both additional ICC devices and water level measurement reliability improvements is not justified by the resulting risk reduction. The risk remaining after inclusion of the water level measurement reliability improvements cited in SLI-8211 is sufficiently small on an absolute basis to preclude the need for further reduction in risk which would be obtained through the use of additional ICC devices. Therefore, the staff agrees with the conclusion drawn in SLI-8218 that if the applicant upgrades the water level system to be consistent with the recommendations cited in SLI-8211, there is no additional instrumentation needed for detecting ICC. Since the Clinton water level instrumentation conforms with the recommendations of SLI-8211, there is no additional instrumentation required for detecting ICC. The staff, therefore, concludes that TMI-2 Action Plan Item II.F.2 related to the instrumentation for detecting ICC is satisfactorily resolved.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.8 Pool Dynamic Loads

In the SER the applicant stated that an evaluation of the Clinton safety/relief valve quencher design as it compares to the Kuosheng design is being performed to determine plant similarity and applicability of the Kuosheng design data to Clinton to eliminate the need for a safety/relief valve (SRV) inplant test. This was identified as part of licensing condition 8 in Section 1.10 of the SER.

In NUREG-0763, "Guidelines for Confirmatory Inplant Test of Safety-Relief Valve Discharge for BWR Plants," the staff states, in part, that inplant tests will be required for those plants in which parameters potentially affecting SRV-discharge performance are deemed to be plant unique. In Section 4 of NUREG-0763, the staff lists five conditions which if satisfied (i.e., if applicants are able to demonstrate that the conditions in their plant are similar to the conditions in plants previously tested), will obviate the need for any new tests.

In its letter dated August 8, 1984, the applicant submitted the requested evaluation and justification. The applicant concluded that inplant SRV testing is not required for Clinton since the Kuosheng SRV test data confirmed the conservative design of the Clinton Mark III containment for SRV hydrodynamic loads. The following summarizes the five conditions in NUREG-0763, the applicant's position regarding these conditions, and the staff's evaluation of the applicant's positions.

Items 1 through 4 of Section 4, NUREG-0763, deal with the quencher geometry and include the important parameters that affect the loads. They are the line length, line area, air volume, quencher submergence, vacuum breaker size, pool area per quencher, quencher location and orientation in the pool, pool geometry, and the steam flow rate.

The applicant provided a dimensional comparison of the quenchers installed at Clinton with those at Perry, Grand Gulf, and Kuosheng (see Table 6.1). Except for those parameters identified by an asterisk, the quenchers are identical.

With respect to those parameters, as identified by analytical methodology, that affect the loads, the applicant provided comparisons which show, except as listed below, that the Clinton-unique parameters do not differ significantly from those tested either at Kuosheng or Grand Gulf. Justification for each parameter that appears to be significantly different from those tested is summarized below.

(1) Safety/Relief Valve Discharge Line Air Volume (DLV) and Length

The DLV is the more critical parameter in the determination of the peak pool pressure; however, as demonstrated in this section, the Clinton DLV variation falls within the recommended range specified in the GE methodology. Therefore, the larger DLV for Clinton will not contribute to any increase in the pool pressure.

The trend of the test data is for the bubble pressure to increase with DLV up to a value of 62.4 ft^3. Beyond this point, the bubble pressure is reduced. Clinton is designed in accordance with GESSAR II methodology, which conservatively assumes the air bubble pressure is maintained at its maximum value despite the physical trends indicating that bubble pressure decreases. The applicant maintains, and the staff agrees, that the Clinton DLV range, which is different from the range tested at Kuosheng or that to be tested at Grand Gulf, is conservatively considered in the approved GESSAR II methodology, and no further testing is required.

(2) Quencher Support

The quencher support method at Kuosheng tends to confine the discharge bubble and to introduce minor variations into the air bubble pressure and frequency. However, since the quencher support methods at Clinton and Grand Gulf are similar, the results from the Grand Gulf test will be directly applicable to Clinton.

On the basis of its review of the applicant's assessments for Items 1-4 of Section 4, NUREG-0763, the staff finds that sufficient similarities exist between the Clinton X-quencher geometry (and the associated plant-unique parameters that affect the SRV hydrodynamic loads) and those tested either at the Kuosheng plant or those that will be tested at the Grand Gulf plant.

With regard to Item 5 of Section 4, NUREG-0763, the applicant has provided comparisons among the structural data of the Mark III containments for the Kuosheng, Perry, Grand Gulf, and Clinton plants. These comparisons indicate that the structural characteristics of the Clinton containment are either similar to or bounded by the Kuosheng and Grand Gulf characteristics with a few minor exceptions. The thickness of the Clinton containment wall is 3 ft in the pool region; thicknesses of the Kuosheng and Grand Gulf containment walls, in the same region, are 8.5 ft and 3.5 ft, respectively. Since in-plant SRV tests are planned for the Grand Gulf containment, the effect of the thickness of the wall in the pool region on the fluid/structure interaction can be easily evaluated. The other minor difference between the structural characteristics of the Clinton containment and the Kuosheng/Grand Gulf containment pertains to the specified concrete compressive strength. The Clinton design is based on a concrete compressive strength of 4,000 psi; the Kuosheng/Grand Gulf containments have been designed with concrete compressive strengths of 5,000 psi. This difference is not expected to have any significant effect on the plant's response to the pool dynamic loadings as the stiffness of the concrete structure is not very sensitive to the compressive strength (stiffness is proportional to the square root of the strength).

Given the above findings, the staff concludes that there exists enough structural similarity between the containments of Clinton, Kuosheng, and Grand Gulf so that the findings and results from the Kuosheng and Grand Gulf in-plant SRV tests should reasonably bound the Clinton responses.

On the basis of its evaluation, the staff concludes that SRV inplant tests are not required for Clinton, since the applicant has demonstrated that the SRV discharge conditions in its plant are sufficiently similar to conditions previously tested at Kuosheng or that will be tested at Grand Gulf. However, as indicated by the applicant in its July 8, 1983, letter, if the Grand Gulf inplant tests identify any specific concerns, the applicant will be required to assess the associated impacts on the Clinton design.

Since the applicability of the Kuosheng SRV quencher design to Clinton has been demonstrated and in Section 3.9.2 of this SSER the staff has accepted the Kuosheng reactor internals test as a valid prototype for Clinton, licensing condition 8 is considered to be resolved.

6.2.7 TMI-2 Requirements

In the SER the staff stated that the applicant complies with the provisions of TMI-2 Action Plan Items II.F.1(4), (5), and (6) pending receipt of confirmatory design details of these monitoring systems.

TMI-2 Action Plan Items II.F.1(4), (5), and (6) require that continuous indication of containment pressure, containment water level, and containment hydrogen concentration be provided in the control room. The applicant is providing instrumentation to monitor these parameters. The design and qualification criteria for this instrumentation are given in RG 1.97 ("Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident"). By letter dated September 9, 1983, the applicant submitted a compliance report for RG 1.97 instrumentation at Clinton, and in FSAR Amendment 26 the applicant provided design information related to these systems. The instrumentation provided to monitor containment pressure, water level, and hydrogen concentration is discussed below.

The prescribed range for containment pressure indication, for steel-lined reinforced concrete containments such as at Clinton, given in TMI-2 Action Plan Item II.F.1(4) and in RG 1.97 is from -5 psig to three times the design pressure of the containment. The design pressure of the Clinton containment is 15 psig. The range of the Clinton containment pressure indication is from -5 psig to 60 psia (45.3 psig), which satisfies the range requirements.

Containment pressure is listed in RG 1.97 as a Category 1 variable. The instrumentation is required to be environmentally and seismically qualified, perform its function given a single failure, be designated as safety related (Class 1E) and subject to the applicable quality assurance criteria, and be continuously displayed in the control room with at least one channel recorded. The applicant has stated that the Clinton containment pressure instrumentation will consist of redundant channels that are environmentally and seismically qualified (in accordance with 10 CFR 50.49 and RG 1.100 respectively), and supplied from Class 1E power sources. Both channels are displayed on recorders in the control room. The containment pressure instrument channels, which use Rosemount 1153 transmitters, have an accuracy of ±0.25% of the calibrated span, and a response time of 0.2 second. On the basis of the staff's review of the applicant's RG 1.97 compliance report, it is concluded that the containment pressure instrumentation at Clinton will satisfy Category 1 requirements, and the requirements of TMI-2 Action Plan Item II.F.1(4) and, therefore, is acceptable.

The prescribed range for containment water level indication for boiling-water reactors (BWRs) is from the bottom of the suppression pool or, for pressure-suppression containments such as Clinton, from the ECCS suction lines, to 5 feet above the normal water level of the suppression pool. Containment water level is listed in RG 1.97 as a Category 1 variable.

The range of the Clinton suppression pool water level instrument channels is from 720 ft to 736 ft. The emergency core cooling system (ECCS) and reactor core isolation cooling (RCIC) suction lines from the suppression pool are located at 720 ft. The normal suppression pool water level is 731 ft 2 in. (19 ft 2 in. above the suppression pool floor). Thus the Clinton suppression pool level indication covers the prescribed range with the exception of 2 in. at the upper end. The staff concludes that the range of the Clinton suppression pool water level indication satisfies the intent of RG 1.97, and therefore, is acceptable. The associated instrumentation consists of redundant channels that are environmentally and seismically qualified, and supplied from Class 1E power sources. Both channels are displayed on recorders in the control room. The suppression pool level instrument channels have an accuracy of ±0.25% of the calibrated span. On the basis of the staff's review of the applicant's RG 1.97 compliance report, it is concluded that the containment water level instrumentation at Clinton will satisfy Category 1 requirements, and the requirements of TMI-2 Action Plan Item II.F.1(5), and therefore, is acceptable.

The prescribed range for containment hydrogen concentration indication is from 0 to 30% of containment atmosphere. For BWRs, control room indication of hydrogen concentration must be provided for both the drywell and the containment. Containment and drywell hydrogen concentration are listed as Category 1 variables in RG 1.97, and must be capable of operating from 12 psia to design pressure.

The range of the Clinton containment and drywell hydrogen concentration indication is from 0 to 30% by volume. There are two redundant divisions of safety-related instrumentation capable of monitoring hydrogen concentration from three zones in the drywell and two zones in the containment. The staff has reviewed the containment and drywell hydrogen sample locations and finds them acceptable. The containment atmosphere monitoring (CAM) system, which performs the hydrogen monitoring function, can be placed into operation within 30 minutes after an accident. The accuracy of the hydrogen monitoring instrumentation is ±1% of full scale. Each division of containment and drywell hydrogen concentration is displayed on a separate indicator. A nonqualified printer records the time, zone, and hydrogen content of each sample. The CAM system is designed to be operable from 12 psia to containment design pressure. The CAM system is further discussed in Section 7.6.3.1 of the SER. On the basis of the staff's review of the applicant's RG 1.97 compliance report, it is concluded that the containment and drywell hydrogen monitoring instrumentation complies with the requirements of RG 1.97 and the requirements of TMI-2 Action Plan Item II.F.1(6) and, therefore, is acceptable.

6.3 Emergency Core Cooling System

6.3.2 Evaluation

6.3.2.3 Functional Design

In the SER the applicant stated that the modification to provide automatic restart of the RCIC system on low water level is being incorporated at Clinton.

The staff has reviewed the information provided in FSAR Amendment 29 and the applicant's February 9, 1984, letter. These submittals document that the RCIC system design at Clinton has been modified in accordance with TMI-2 Action Plan Item II.K.3.13 to automatically restart if vessel level should fall to the low level initiation setpoint (level 2) following automatic termination on vessel high level (level 8). The level 8 trip function has been moved from the RCIC turbine trip valve to the RCIC turbine steam supply valve. The steam supply valve (1E51-F045) will automatically open on an RCIC initiation signal (automatic on level 2, 1-out-of-2-taken-twice logic; or manual) if valve 1E51-F068, RCIC turbine exhaust to the suppression pool, is open. If 1E51-F068 is not fully open (this valve is normally open), "RCIC OUT OF SERVICE" annunciation is received in the control room. Following initiation, if vessel level should increase to level 8, 1E51-F045 will automatically close, terminating RCIC flow to the reactor vessel. A 2-out-of-2 logic (trip modules B21-N693B, Division 2 power; and B21-N693A, Division 1 power) is used for level 8 termination. If reactor vessel water level falls below level 8, the RCIC termination signal will clear, and 1E51-F045 will remain closed. The level 8 signal will clear when either level 8 instrument channel senses that vessel level is less than setpoint. Thus, no single failure of a level 8 channel can prevent RCIC system operation because of a spurious level 8 termination signal, or because the termination signal fails to clear. When the steam supply valve, 1E51-F045, closes on level 8, the RCIC system is automatically placed in a standby configuration. If reactor vessel water level should fall to the low level initiation setpoint, 1E51-F045 will reopen, again providing RCIC flow to the vessel.

The staff concludes that the Clinton RCIC system design conforms to the requirements of TMI-2 Action Plan Item II.K.3.13 regarding automatic restart, and, therefore, is acceptable.

As stated in the SER, the staff requires installation of this modification before fuel loading or, if qualified equipment is not available before fuel loading, the staff requires installation during the first refueling outage of sufficient duration after qualified equipment is available.

Table 6.1 X-quencher comparison

Parameter	Kuosheng	Perry	Grand Gulf	Clinton
Reducer length, ft*	1.667	2.813	2.417	1.646
Hub length, ft*	3.229	2.00	2.00	2.00
Bottom cap length, ft*	1.0	0.85	0.85	0.85
Hub to end of arm, ft	4.875	4.875	4.875	4.875
Hub to first row of holes, ft	1.896	1.890	1.896	1.885
Length of hole pattern, ft	2.625	2.624	2.625	2.625
Hub diameter, in.	24, Sch. 80	24, Sch. 140	24, Sch. 120	24, Sch. 120
SRVDL diameter, in.	10, Sch. 80	10, Sch. 40S	10, Sch. 80	10, Sch. 40S
Arm diameter, in.	12, Sch. 80	12, Sch. 80	12, Sch. 80	12, Sch. 80
Reducer taper, degrees*	17.1	10.75	10.4	10.75
Angle between arms, degrees	80-80-80-120	80-80-80-120	80-80-80-120	80-80-80-120
SRVDL air volume, ft^3/SRVDL	42.7/74.4	55.7/107.3	56.8/76.6	67.1/103.0
Line length, ft	47.7/82.4 46.0/79.6	44.9/82.6		56.9/91.3
Pool area/quencher, ft^2	332	310	333	448
Submergence, ft	13.8	14.0	13.8	13.9
Vacuum breakers, 2 lines, in.	10	6	10	10

*Quenchers identical except for these parameters.

7 INSTRUMENTATION AND CONTROLS

7.3 Engineered Safety Features Systems

7.3.3 Resolution of Issues

7.3.3.3 Modify Break Detection Logic To Prevent Spurious Isolation of High-Pressure Coolant Injection and Reactor Core Isolation Cooling System (TMI-2 Action Plan Item II.K.3.15)

In the SER the staff stated that the conceptual design provided by the applicant regarding circuit modifications to prevent spurious RCIC system isolation from the RCIC steam supply line break detection circuitry was acceptable.

The staff has reviewed the information provided in FSAR Amendment 29 and the applicant's February 9, 1984, letter. These submittals document that the RCIC system design at Clinton has been modified in accordance with (1) TMI-2 Action Plan Item II.K.3.15 to prevent spurious isolation on system startup and (2) the conceptual design previously approved by the staff. Four differential pressure transmitters (1E31-N083A&B and 1E31-N084A&B), one located at each of four elbows in the RCIC turbine steam supply line, are provided to detect a downstream break. High differential pressure resulting from high flow in the line because of a break will initiate closure of the inboard and outboard RCIC steam supply line isolation valves 1E51-F063 and 1E51-F064. However, pressure spikes which occur on system startup have resulted in spurious RCIC system isolations. The applicant has modified the pipe break detection circuitry to include 3-second solid state time delays in the isolation logic to prevent RCIC isolation from short duration pressure spikes on system startup. The timers are started when differential pressure exceeds the trip setpoint. At the end of the 3-second period, isolation will occur only if steamline differential pressure remains above setpoint. The timers automatically reset when differential pressure returns below setpoint. The applicant has stated that for postulated system pipe breaks, releases that result from a 3-second delay will be less than the design-basis conditions, and within existing safety analyses. The staff will require that the Clinton Technical Specifications contain provisions for periodic surveillance and calibration of the 3-second timers.

The staff concludes that the above modification will ensure that RCIC isolation is based on continuous high steam flow indicative of a break, and will prevent pressure spikes that occur on RCIC initiation from causing inadvertent isolation. The Clinton RCIC design conforms to the requirements of TMI-2 Action Plan Item II.K.3.15 regarding spurious isolation and, therefore, is acceptable.

As stated in the SER the staff requires installation of this modification before fuel loading or, if qualified equipment is not available before fuel loading, the staff requires installation during the first outage of sufficient duration after qualified equipment is available.

7.3.3.4 Modification of Automatic Depressurization System Logic - Feasibility for Increased Diversity for Some Event Sequences (TMI-2 Action Plan Item II.K.3.18)

The SER stated that the applicant will submit the design details of one of the two alternatives which have been accepted by the staff to resolve this item and verify that the automatic depressurization system (ADS) logic modification will be implemented before fuel loading.

The staff has reviewed the information provided in FSAR Amendments 27 and 29 and the applicant's February 9, 1984, letter. These submittals document that the ADS design has been modified in accordance with TMI-2 Action Plan Item II.K.3.18 to initiate automatically in the absence of a high drywell pressure initiation signal. The ADS functions as a backup to the high-pressure core-spray (HPCS) system by depressurizing the reactor vessel so that low-pressure systems may inject water for core cooling. The ADS is typically actuated upon coincident signals of reactor vessel low water level, high drywell pressure, a low-pressure ECCS pump running, and a 105-second delay which allows ADS to be bypassed if the operator believes the actuation signal is erroneous or if vessel water level can be restored. However, for transient and accident events which do not produce high drywell pressure, and are further degraded by a loss of HPCS, manual actuation of the ADS would be required to ensure adequate core cooling.

In order to eliminate the need for manual ADS actuation to ensure adequate core cooling, the applicant has installed bypass timers which will automatically bypass the drywell high-presssure inputs required for ADS actuation if reactor vessel water level remains below the ADS initiation setpoint (level 1) for a sustained period (6 minutes). After the 6-minute delay and the 105-second delay, ADS will be automatically actuated in the absence of a drywell high-pressure signal if a reactor vessel low-water-level condition still exists and a low-pressure ECCS pump is running. Annunciation is provided in the control room when the 105-second timers are initiated. Annunciation is also provided when a reactor vessel low-water-level or drywell high-pressure condition is detected.

In response to a request from the staff the applicant, by letter dated February 1, 1985, submitted supplemental information in support of the proposed 6-minute time delay. In this submittal, the applicant stated that the nominal time delay of 6 minutes was chosen to be consistent with analyses of plant behavior during limiting transients and that detailed analyses were performed to ensure: (1) the avoidance of excessive fuel cladding heatup using the 10 CFR 50, Appendix K, models for the most limiting transient described in FSAR Section 6.3, and (2) that sufficient time is provided to allow recovery of RPV water level above level 1 during an ATWS event.

A 2.66-ft^2 steamline break occurring outside of containment was identified as the most limiting transient associated with determining an appropriate bypass timer delay (Figure 6.3-72 of the Clinton FSAR). The analysis used the Appendix K models for calculating the peak cladding temperature response assuming an ADS bypass timer delay of 7 minutes with no operator action and failure of HPCS. The calculated peak cladding temperature for this case was about 1600°F compared with the limiting value of 2200°F. The use of the cited analysis with the 7-minute delay time is conservative since a 6-minute delay would result in an earlier injection of ECCS (LPCI) coolant with a corresponding lower core heatup.

Four 6-minute delays have been added, one for each ADS drywell high-pressure initiation channel. There are two ADS actuation channels (Division 1 and Division 2), either of which can perform the required ADS function. There are two bypass timers associated with each ADS division. The staff will require that the Clinton Technical Specifications contain provisions for periodic surveillance and calibration of the 6-minute bypass timers and the 105-second timers. The 6-minute timers automatically reset when vessel level increases above level 1, and the 105-second timers automatically reset at the end of the 105-second period.

Another modification made to the Clinton ADS consists of the addition of two ADS inhibit switches (one per ADS division) that permit the operator to override the ADS automatic blowdown logic if necessary. These manual inhibit switches prevent automatic ADS actuation, but do not inhibit the safety-relief valve (SRV) pressure-relief function, manual ADS actuation, or individual SRV control. The applicant has stated that addition of the ADS manual inhibit switches will simplify the execution of those steps in the emergency procedure guidelines (EPGs) related to ATWS mitigation. The inhibit switches are two-position (NORMAL and INOP), maintained-contact switches. Placing a switch in the INOP position, which defeats the ADS automatic actuation logic for the associated division, causes "ADS OUT OF SERVICE" annunciation in the control room for that division, and actuates an "ADS INHIBITED" status light on control room panel H13-P601.

The staff concludes that the Clinton ADS design including the basis for the 6-minute ADS time delay conforms to the requirements of TMI-2 Action Plan Item II.K.3.18 regarding ADS automatic actuation to ensure adequate core cooling and, therefore, is acceptable.

As stated in the SER, the staff requires installation of this modification before fuel loading, or if qualified equipment is not available before fuel loading, the staff requires installation during the first outage of sufficient duration after qualified equipment is available.

7.6 Interlock Systems Important to Safety

7.6.3 Resolution of Issues

7.6.3.1 Containment Atmosphere Monitoring (CAM) System

In the SER the staff stated that the applicant will provide a detailed design description of the CAM system. The staff has reviewed the information provided in FSAR Amendment 26. This amendment documents that the CAM system design consists of two redundant and independent divisions of safety-related equipment used to monitor normal and postaccident hydrogen and oxygen concentrations, and high-range gamma radiation levels, in both the drywell and containment. The CAM system is designed as Class 1E, seismic Category I, and to operate in a postaccident environment. The CAM system instrumentation is powered from the divisional buses, and is energized during normal operation, shutdown, and following an accident. The CAM system is used for control room indication only (inputs are provided to indicators, recorders, status lights, the computer, and the main annunciator system); there are no associated control or actuation functions. The applicant has stated that no single failure will cause the loss of indication of hydrogen, oxygen, or gamma radiation, and that the CAM system design complies with the requirements of RG 1.97. The CAM system can be tested during operation.

On the basis of the information provided, the staff concludes that the Clinton CAM system design conforms to the applicable requirements of Section 7.6 of the Standard Review Plan and, therefore is acceptable.

The staff requires installation of this modification before fuel loading, or if qualified equipment is not available before fuel loading, the staff requires installation during the first outage of sufficient duration after qualified equipment is available.

8 ELECTRIC POWER SYSTEM

8.3 Onsite Emergency Power Systems

8.3.1 AC Power System

In its original evaluation, the staff stated that because the Clinton Division 3 (HPCS) diesel generator is different from the diesel generator combination tested and reported in Amendment 3 of GE Topical Report NEDO-10905, it must undergo similar prototype qualification testing described in that report. The staff reported that the applicant had committed to perform this testing. The subject testing consists of 69 starts of the HPCS diesel generator followed by loading of the entire complement of HPCS loads in the HPCS full loop configuration. This simulates the actual loading that would be seen on the HPCS diesel generator following a LOCA together with a loss of offsite power event.

In subsequent letters dated February 15, 1983; October 14, 1983, and June 13, 1984, the applicant requested that the requirement for testing in the full loop configuration be reduced on the basis that 69 full system starts would impose undue stress and wear on the system, and that similar versions of the Clinton HPCS diesel generators had already demonstrated their reliability. The modified testing program proposed by the applicant consists of 8 full loop tests, and 61 additional HPCS diesel generator starts followed by loading to the grid at approximately the HPCS pump motor load.

To support its request for a modified test, the applicant submitted starting reliability test data (letters dated Aug. 6, 1984, and Oct. 22, 1984) on diesel generator units where at least one of the diesels (some were tandem configuration) was a 16-cylinder model EMD-645E4 engine, the same as the diesel in the Clinton HPCS design. Although these reports are not all directly applicable to the diesel generator configuration and application in the HPCS system at Clinton they do provide sufficient general base data of the Clinton HPCS diesel engine reliability so as to warrant the proposed reduced testing. The proposed tests will still be conducted in the full system configuration but only for 8 starts in lieu of the full 69. This will, therefore, still provide a measure of the machine's capability to start, and assume its full system load. The remaining 61 starts loaded to the grid will provide a measure of the machine's ability to carry its required load. Thus, the combination of these tests will demonstrate the machine's capability to start and come up to synchronous speed and rated voltage in the required time interval and accept and carry full system load.

The modified 69-start test proposed by the applicant is, therefore, acceptable. The other preoperational tests outlined in Regulatory Guide (RG) 1.108 should still be performed as prescribed.

12 RADIATION PROTECTION

12.3 Radiation Protection Design Features

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

The SER contained a conditional acceptance of the high range containment monitoring system for Clinton Power Station Unit 1, subject to the applicant providing plant layout drawings showing the location of the four detectors. The necessary information was provided in the applicant's January 14, 1985, letter. The staff has reviewed the proposed locations of the four detectors and concluded they meet the intent of TMI-2 Action Plan Item II.F.1(3) and, therefore, are acceptable.

13 CONDUCT OF OPERATIONS

13.3 Emergency Preparedness Evaluation

In previous supplements to the Safety Evaluation Report (SER), the staff has reported its findings on the adequacy of emergency planning for the Clinton Power Station on the basis of the staff's review of the applicant's original post-TMI emergency plan and the first revision to that plan. Since that first revision, the applicant has continued to revise and upgrade its emergency plan and has issued three additional substantive revisions.

Because of the substantive nature of the several revisions to the applicant's plan, the staff has completely revised and updated its evaluation of the applicant's emergency plan. That new evaluation is presented in Sections 13.3.1 through 13.3.3 (this supplement), and supersedes previous staff evaluations. All previously identified unresolved emergency planning issues have been resolved.

13.3.1 Introduction

The staff's evaluation of the state of emergency preparedness associated with the Clinton Power Station involves review of the applicant's onsite emergency plans and preparedness, as well as review of the Federal Emergency Management Agency (FEMA) findings and determination pertaining to the adequacy of offsite (State and local) emergency plans and preparedness.

In September 1981 (FSAR Amendment 7), the applicant, Illinois Power Company et al. (IPC), filed with the NRC its original emergency plan to meet the revised emergency planning requirements of 10 CFR 50. Since that time, the applicant has continued to upgrade its emergency plan based upon the installation and checkout of equipment and systems; refinements identified as part of its coordination with offsite authorities during the development of offsite planning; and continued discussions with the NRC staff. The applicant has submitted four revisions to its emergency plans which were filed with the NRC as follows: Revision 1, FSAR Amendment 16, May 1982; Revision 2, FSAR Amendment 28, December 1983, Revision 3, FSAR Amendment 31, October 1984; and Revision 4 (to be incorporated into the FSAR), December 1984. The staff reviewed each of those revisions in detail and, where additional clarification was needed, has discussed those matters with the applicant. The staff has completed its review and evaluation of the adequacy of the latest revised emergency plan (Revision 4) and the results of that evaluation are provided here.

The acceptance criteria used as the basis for the staff's review of the applicant's emergency plan are specified in Section 13.3, "Emergency Planning," of the Standard Review Plan (SRP), NUREG-0800, dated July 1981, and include: the planning standards of 10 CFR 50.47(b); the requirements of Appendix E to 10 CFR 50; and the specific guidance criteria of NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 1980. The guidance criteria of NUREG-0654 have been endorsed in

Regulatory Guide (RG) 1.101, Revision 2, "Emergency Planning and Preparedness for Nuclear Power Reactors," dated October 1981 and thus have the same status as a regulatory guide.

The SRP states that FEMA findings and determinations on the adequacy of offsite plans will be reviewed by the NRC. Interim findings and determinations by FEMA on the adequacy of the State and local county emergency plans have not yet been received as the local plan is still under development. The local plan is expected to be filed with FEMA during the first quarter of 1985 and interim findings are expected during the second quarter of 1985. FEMA has already reviewed the State of Illinois emergency plan and determined that it is adequate. That review was performed in connection with other operating nuclear power plants in Illinois. The State of Illinois plan is generic in nature and site-specific provisions are found in the local plan for each nuclear power plant (which is still under development for Clinton). In addition, a full-participation exercise is scheduled for September 1985 and the staff will report the results of that exercise and the FEMA finding on the offsite plans in a future SER supplement.

This evaluation of the applicant's emergency plan follows the format of Part II of NUREG-0654 and the 16 specific planning standards of 10 CFR 50.47(b). Each of the planning standards is listed and followed by a summary of applicable portions of the emergency plan that relate principally to that specific standard. The summary conclusions of the staff's review of the applicant's emergency plan are provided in Section 13.3.3.1, this supplement.

13.3.2 Evaluation of Applicant's Emergency Plan - Findings on Standards and Criteria

13.3.2.1 Assignment of Responsibility (Organizational Control)

Planning Standard

> Primary responsibility for emergency response by the nuclear facility licensee and by State and local organizations within the emergency planning zones (EPZs) have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

Evaluation

The Federal, State, and local organizations that are intended to be part of the overall response organization for the EPZs are identified. The role of the State of Illinois is fully described, with reference to the State of Illinois Plan for Radiological Accidents (IPRA) and the IPRA site-specific Annex for Clinton (under development).

The applicant's concept of operations and its relationship to the total emergency response effort is described and block diagrams showing the interfaces between and among the principal response organizations are provided.

Written agreements are included to verify assistance arrangements between the plan and other support organizations to provide for radiological support, medical assistance, medical transportation, and fire protection during an emergency. The applicant has also committed to update all letters of agreement before fuel load to ensure all information is current.

The emergency plan identifies the specific individual(s), by title, who will be in charge of the applicant's emergency response and specifies the functions, responsibilities, and authorities of key individuals. More detailed discussions of emergency duties and responsibilities are found in the emergency plan implementing procedure (EPIP) EC-01, "CPS Emergency Response Organizations and Staffing."

There is a 24-hour/day communication linkage capability between the facility and Federal, State, and local response agencies and organizations to ensure rapid transmittal of accurate notification information and emergency assessment data.

The emergency plan provides a description of the onsite and offsite organizations for continuous (24-hour) operation for a protracted period. The description covers both personnel aspects and equipment aspects of protracted responses. The Administrative Supervisor is identified as the individual responsible for assuring continuity of resources.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.2 Onsite Emergency Organization

Planning Standard

> On-shift facility licensee responsibilities for emergency response are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite response activities and offsite support and response activities are specified.

Evaluation

The applicant's emergency plan describes the onsite emergency organization of plant personnel for all shifts and its relation to the responsibilities and duties of the normal shift complement (Table 2-1 of the emergency plan). Positions and/or titles of shift and plant personnel (both onsite and offsite) assigned emergency functional duties are listed. The shift and augmented staffing specified in the emergency plan meet the specific staffing goals expressed in Table B-1 of NUREG-0654. The emergency plan identifies the Emergency Manager (initially the Shift Supervisor) as the individual who has the responsibility and authority for continued evaluation, coordination, and control of all onsite activities related to an emergency and establishes a specific line of succession for the Emergency Manager (EM) position. The plan also identifies the specific criteria (e.g., methods and procedures) by which the EM position will be transferred. The Emergency Manager may not delegate his responsibility for recommendations to the offsite authorities concerning evacuation or other protective actions.

The emergency plan identifies four emergency positions as having the "Command Authority" (as defined in Sections 1.5.1 and 2.5.1 of the plan) during a response to an emergency. These four positions actually exist only one at a time so that there is only one individual directing emergency response at any given time. When an emergency condition arises, the Shift Supervisor will be designated as the Interim Station Emergency Director and will operate out of the main control room. It will be the Shift Supervisor's responsibility, as the Interim Station Emergency Director, to evaluate the situation. If, in the Shift Supervisor's judgment, conditions meet or exceed any of the emergency classification action levels, it will be the Shift Supervisor's responsibility to implement the emergency plan. Once the emergency plan is implemented, the applicant's emergency response organization will expand, as necessary, to provide adequate management and support personnel to effectively respond to the specific accident conditions and response needs. As the applicant's emergency response organization expands, the Command Authority will shift (1) from the Shift Supervisor to the Station Emergency Director operating from the Technical Support Center (TSC); (2) then from the Station Emergency Director to the Emergency Manager operating from the Emergency Operation Facility (EOF); and ultimately (3) from the Emergency Manager to the Recovery Manager operating from the EOF for Recovery Operations.

The interfaces between and among the onsite functional areas of emergency activity, licensee headquarters support, local support services, and State and local government response organizations are specified (Figure 2-9 of the emergency plan). The corporate management, administrative, and technical support personnel who will augment the plant staff, is specified in Figure 2-8 of the emergency plan and personnel assigned responsibilities in the areas of logistical and technical support are identified. Public information will be coordinated by the IPC Public Information Officer, who is official company spokesperson (IPC Executive Vice President or his designated alternate). Logistics support is the responsibility of the Administrative Supervisor.

Contractors and private organizations who may be requested to provide technical assistance to and augmentation of the applicant's emergency organization are specified. Police, ambulance, medical, hospital, and fire-fighting support which can be provided by local agencies is identified.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.3 Emergency Response Support and Resources

Planning Standard

> Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate State and

local staff at the licensee's nearsite Emergency Operations Facility have been made, and other organizations capable of augmenting the planned response have been identified.

Evaluation

The applicant's emergency plan identifies the Emergency Manager (or equivalent position) as being authorized to request Federal (U.S. Department of Energy) assistance. The plan also identifies the expected Federal response resources and the expected time of arrival of the assistance. The emergency plan identifies the TSC Technical Assessment Supervisor and the TSC and EOF Administrative Supervisors as being responsible for the direction and coordination of the Federal assistance effort. The applicant has also committed to review the various time and manpower requirements necessary for effective coordination of Federal responses and may upgrade its emergency response organization (as necessary) to provide such additional manpower (as necessary) to ensure proper use of all available response resources. A letter of agreement with the U.S. Department of Energy pertaining to the Federal response has been included in the plan.

Provisions have been made by the applicant to dispatch a representative of Illinois Power Company to the local Emergency Operations Centers for DeWitt County (in Clinton, Illinois) and to the State of Illinois (in Springfield, Illinois).

The emergency plan identifies the mobile laboratories of the U.S. Department of Energy and the Illinois Department of Nuclear Safety, as having the capability for backup radiological analyses should the normal plant laboratory become unavailable as a result of radiological contamination or high background radiation during an emergency. The staff concludes that the capability of the mobile laboratories identified by the applicant would be adequate for processing and analyzing environmental samples. The emergency plan also identifies the Fermi National Accelerator Laboratory and the Argonne National Laboratory of the U.S. Department of Energy, which could be used for processing and analyzing post-accident samples of primary coolant or containment atmosphere, should the site's laboratories become unavailable during an emergency.

In addition, the emergency plan identifies emergency services that could be provided by other organizations. Technical assistance may be requested from the Clinton architect-engineer, Sargent & Lundy, and from the nuclear steam system supplier, General Electric Co., as well as from suppliers of various equipment used on site. The Institute of Nuclear Power Operations (INPO) also administers an industrywide mutual aid agreement for technical assistance to which IPC is a party.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.4 Emergency Classification System

Planning Standard

A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and State and local response plans call for reliance on information provided by facility licensees for determinations of minimum initial offsite response measures.

Evaluation

The applicant's emergency plan establishes an emergency classification scheme in accordance with that set forth in Appendix 1 to NUREG-0654 (RG 1.101, Rev. 2). The four classes of emergencies are: Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency.

Observable and measurable emergency action levels (EALs) have been established which, if exceeded, will initiate each emergency class, consistent with the criteria of Appendix 1 to NUREG-0654. These EALs are composed of a combination of plant parameters (such as instrument readings and system status) that can be used to give a relatively quick indication of the severity of the accident.

The applicant has identified plant system and effluent parameters and annunciators characteristic of a spectrum of offnormal conditions and accidents. These parameters have been used to develop specific initiating conditions (ICs) which have been divided into 10 broad sections:

(1) emergency core cooling systems
(2) radiation monitoring
(3) abnormal temperatures
(4) failure of safety system
(5) control power
(6) fires
(7) security
(8) natural phenomena
(9) Technical Specifications
(10) other hazardous conditions

Specific EALs will be described in the appropriate plant emergency plan implementing procedures. Those procedures will contain specific information and guidance for determining the appropriate EAL and properly classifying the emergency condition, as well as the appropriate response actions to be taken.

Finding

The staff has reviewed the proposed example EALs and ICs as presented in the applicant's emergency plan (Tables 4-1 through 4-4) and implementing procedure EC-02, "Emergency Classification." The staff has determined that applicant's EAL and IC schemes meet the intent of this planning standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654. The staff notes that many of the initiating conditions refer to the specific parameter values being developed in the Technical Specifications for Clinton and are, therefore, not yet in final form. The staff also noted that the EAL information provided in the applicant's emergency plan are examples only and that the exact and specific nature of the initiating condition appears in the emergency plan implementing procedures.

The staff will confirm the adequacy of the applicant's final EAL and IC schemes as part of its onsite implementation appraisal of the applicant's emergency preparedness program.

13.3.2.5 Notification Methods and Procedures

Planning Standard

> Procedures have been established for notification by the licensee, of State and local response organizations and for notification of emergency personnel by all response organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instruction to the populace within the plume exposure pathway Emergency Planning Zone have been established.

Evaluation

Procedures that describe mutually agreed-upon bases for notification of response organizations consistent with the emergency classification scheme, including means for verification, are established in the applicant's emergency plan. Provisions also are established for alerting, notifying, and mobilizing emergency response personnel. Onshift personnel will be notified by the plant public address system, and offduty and corporate personnel can be notified by either telephone or the companywide radio and pocket-paging systems. Communications off site may be made by one or more of the following means: conventional telephone system/radio system, private telephone lines, dedicated telephone lines, and emergency radio systems.

The contents of the initial emergency message to be sent from the plant to offsite authorities are preestablished. Initial notification will be made within 15 minutes of emergency declaration, regardless of the classificaton. Followup messages containing appropriate information from the facility to offsite authorities are established. The licensee will assist the State and local organizations in preparing written messages (i.e., containing information with regard to specific protective actions) intended for the public.

Section 3.2.6 of the emergency plan describes the Clinton Power Station (CPS) Alert and Notification Systems (ANS). The applicant has installed high-powered, fixed, pole-mounted sirens to cover the entire 10-mile plume exposure pathway EPZ. The alert system (sirens) can be activated from either the DeWitt County Sheriff's Office or the Clinton Fire Station. The applicant will notify offsite authorities and will provide recommendations for protective actions for the public using special dedicated telephone systems. The Statewide Nuclear Accident Reporting System (NARS) will be the primary system used for such notifications and recommendation to State and local authorities. The NARS is manned on a 24-hour/day basis and adequate backup communications capabilities exist should the NARS circuit be unavailable for any reason. The emergency plan identifies the Governor of Illinois, the Director of the Illinois Emergency Services and Disaster Agency (ESDA), the Director of the DeWitt County ESDA, and the DeWitt

County Sheriff as having the authority to authorize activation of the public alert and notification system. The emergency plan also references sample Emergency Broadcast System (EBS) messages found in Chapter 4 of the Illinois Plan for Radiological Accidents. The staff has determined that the administrative and physical means proposed by the applicant for notifying and providing prompt instructions to the public within the plume exposure pathway EPZ appear adequate to meet the requirements of 10 CFR 50.47(b)(5) and Section IV.D of 10 CFR 50, Appendix E, for licensing. The final determination of the overall adequacy of the installed warning system, in accordance with Appendix 3 of NUREG-0654, will be made by FEMA as part of FEMA's formal 44 CFR 350 process.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.6 Emergency Communications

Planning Standard

> Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

Evaluation

The applicant's emergency plan provides for primary and backup communication links with the Federal, State, and local emergency response organizations. Provisions exist for 24-hour/day notification to, and activation of, these organizations. The Clinton communication system is designed to provide reliable, redundant, and diverse communications to all essential onsite and offsite locations during normal operation and under accident conditions. The plant's normal communications system includes a public address system, pocket-pager system, dial telephone system, microwave system, sound-powered telephone system, and intraplant two-way radio system. Offsite communications systems include commercial telephone systems, private telephone lines, dedicated emergency telephone lines, general radio systems, and special radio systems developed for emergency use only.

Communications with contiguous State/local governments within the EPZs are provided and will be tested monthly. Communications with Federal response organizations, the State and local emergency operations centers, and field monitoring teams will be tested annually as part of the communication drills. Organizational titles and alternates for both ends of the communication links are given. Communications with Federal emergency response organizations are provided as needed.

Communications between the nuclear facility and the EOF, State and local EOCs, and radiological monitoring teams are provided. The alerting and activating of emergency personnel is provided for. Communication with NRC Headquarters and the NRC Regional Office is also provided.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.7 Public Education and Information

Planning Standard

> Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with the news media for dissemination of information during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.

Evaluation

The emergency plan (Section 2.6.2) describes provisions for public information and education. The applicant, in cooperation with rural electric cooperatives serving the area around the Clinton Power Station, has developed a computer listing of residents within that area. A coordinated yearly dissemination of information to the public regarding how the public will be notified and what their actions should be in an emergency is being developed. The public education information will be distributed by various means and may include periodic information notices enclosed in utility bills, posting of information in public areas, and distribution of publications on an annual basis.

Provisions are being made for written material that is likely to be available in a residence during an emergency and for written material that is likely to be available to any transient population.

During emergency situations, the IPC Executive Vice President, or a qualified alternate, is responsible for coordinating all information releases to the news media and acts as the official company spokesperson. In that capacity, he will disseminate initial and followup information through the news media by means of periodic press releases at the Joint Public Information Center (JPIC) in Decatur, Illinois. A Rumor Control Team is also located in the JPIC to answer telephone inquiries.

The applicant will offer annual training for news media personnel to acquaint them with the emergency plan, to give them information about radiation, and to give them points of contact for release of public information during an emergency.

The applicant's emergency plan describes the procedures for preparing and distributing press releases and for conducting news conferences. Provisions are also made to permit press tours of the CPS Visitor's Center and to provide the press a location for using photographic and other video equipment if radiological conditions allow.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654. The staff will confirm the acceptability and distribution of the applicant's final public information material as part of the preoperational inspection process.

13.3.2.8 Emergency Facilities and Equipment

Planning Standard

> Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

Evaluation

The emergency facilities needed to support an emergency response are provided, including a Technical Support Center (TSC), Emergency Operations Facility (EOF), and Operations Support Center (OSC). The plan provides pertinent information regarding the location, structure, habitability, staffing, communication systems, instrumentation, and records availability at onsite emergency centers.

The emergency plan and emergency response facilities (ERFs) provide for a Technical Support Center which is separate from the control room located adjacent to it and originally designated as the Unit 2 control room. This location allows easy and timely access to the control room and inherently meets the habitability criteria for a TSC.

The applicant has the capability to display and transmit data and data summaries describing plant status to the control room, the TSC, and the EOF. There is space in the TSC for management and technical personnel to perform their functions. The radiological habitability of the TSC is the same as the control room and communications are provided between the control room, the OSC, the EOF, the NRC, and other offsite agencies.

The TSC is capable of supporting reactor control functions, evaluating and diagnosing plant conditions, and serving as the main communications link between the control room, the OSC, the EOF, and the NRC. The TSC can carry out the EOF functions until the EOF is staffed and fully operational.

The OSC is adjacent to the TSC and essentially consists of the balance of the area originally designated as the Unit 2 control room. The OSC provides a place where operations support personnel can assemble and report in an emergency and can receive instructions from the operating staff. The OSC has communications with the control room, the TSC, and the EOF. Both the TSC and the OSC are activated for an Alert or higher emergency classification.

The primary EOF is located within the Clinton Power Station security fence about 1,000 ft east of the control room. It will be used to evaluate and coordinate emergency response operations on a continuing basis by the applicant, as well as by Federal, State, and local officials. It will also be the center for coordinating field-monitoring information, for collecting field samples, and for providing other technical information including recommendations for offsite protective actions. Provisions are made to accommodate representatives from Federal, State, and local government organizations and contractor and other support groups.

Space is provided in the EOF for management and technical personnel to perform their functions. There are communications links between the EOF and the control room, the TSC, the OSC, the NRC, and other offsite agencies. The EOF appears to be capable of supporting the coordination of all the applicant's onsite and offsite activities for response to emergency situations. The EOF will be activated for a Site Area or General Emergency.

The applicant also provides for a backup EOF at IPC's Electric and Gas Dispatch Center in Decatur, Illinois, should the primary EOF become radiologically uninhabitable. This location places the backup EOF about 21.7 miles from the main control room and Technical Support Center.

Onsite monitoring systems are described; such systems include geophysical phenomena monitors (meteorological, hydrological, and seismic), radiological and process monitors, and fire and combustion product detectors. Arrangements have been made to acquire data from or have emergency access to offsite monitoring and analysis equipment including meteorological information from the U.S. National Weather Service.

The applicant will maintain a number of emergency kits that contain protective equipment, communications equipment, radiological monitoring supplies, and other emergency supplies. Provisions are made to inventory, inspect, and check these kits at least once each calendar quarter and after each use. There are enough reserve items to replace equipment removed from kits for maintenance or calibration.

Finding

The staff finds that at the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654 on an interim basis for licensing. The staff will confirm the adequacy of the applicant's final ERFs during a postimplementation inspection in accordance with the requirements of Supplement 1 of NUREG-0737 on a schedule to be developed between the applicant and the NRC.

13.3.2.9 Accident Assessment

Planning Standard

> Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

Evaluation

Although the plant system effluent parameter values for off-normal conditions are identified in Section 4 of the applicant's emergency plan, the applicant has not yet finalized the exact nature of the EALs (see previous discussion of the planning standard on the emergency classification system (Section 13.3.2.4, above) but EALs will be issued in final form in the Emergency Plan Implementing Procedures.

The postaccident sampling capabilities as required by NUREG-0737 for high range monitoring of effluent and containment radiation levels are shown on system parameter tables (see Tables 3-3 and 3-5 through 3-7 of the emergency plan).

The emergency plan describes a radiation monitoring system (RMS), which includes the area RMS, the process RMS, and centralized digital processing, annunciating, and control equipment. Using information from this system, the applicant can determine the source term and magnitude of releases of radioactive materials by use of the RMS computer. The system is also capable of calculating the relationship between effluent monitor readings and onsite and offsite exposures and contamination for various meteorological conditions. A postaccident sampling system will also be installed to allow sampling of the primary coolant, suppression pool water, drywell and containment sumps, drywell and containment atmospheres, and effluent from the reactor water cleanup system while limiting radiation exposure to operating personnel.

The emergency plan describes the capability and resources to acquire and evaluate meteorological data that meet Appendix 2, NUREG-0654, criteria. These data will be available on the RMS cathode-ray tube terminal in the main control room, the Technical Support Center, and the Emergency Operations Facility.

Methodologies and procedures are available for determining source terms of releases, release rates, and projected doses even if the instrumentation used for assessment is offscale or inoperable. Capabilities and resources for field monitoring within the plume emergency planning zone are described, including the capability to measure radioiodine concentrations of 10^{-7} Ci/cc under field conditions. The means for relating measured field contamination levels to dose rates will be provided in the implementing procedures.

Dose rate information may be used in conjunction with dose projections generated by the RMS computer to develop a picture of actual plume travel, integrated dose, and dose rate estimates for key isotopes and for the total radioactivity released.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.10 Protective Response

Planning Standard

> A range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

Evaluation

The emergency plan establishes a range of onsite protective actions including evacuation, distribution of radioprotective drugs, and the use of respiratory protection. Monitoring and decontamination of onsite evacuees will be conducted at the designated assembly area. The storage locations for emergency equipment and supplies are specified in the emergency plan.

The emergency plan describes the means (station alarms and station public address system) to notify the onsite individuals and individuals within the owner-controlled area of the emergency condition.

Accountability of all onsite individuals within 30 minutes and continuous accountability thereafter are specified.

The emergency plan provides for the prompt notification and recommendation of protective actions to State and local authorities for the population at risk in the plume exposure pathway EPZ. Those recommendations are based in part on the U.S. Environmental Protection Agency's Emergency and Lifesaving Protective Action Guides (PAGs). Time estimates for evacuation within the plume exposure pathway EPZ are provided in the applicant's draft Evacuation Time Estimates (July 1984). Population distribution by sector and distance within the plume exposure EPZ is compiled and is included in the plan (see Figure 4-4 in the applicant's emergency plan).

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.11 Radiological Exposure Control

Planning Standard

> Means of controlling radiological exposures, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

Evaluation

Onsite exposure guidelines consistent with the EPA PAGs are established for removing injured personnel, for corrective and assessment action first aid, personnel decontamination, medical transport, and medical treatment services. Provisions exist for 24-hour/day capability to determine the exposures of emergency personnel involved in the response to an accident, for appropriate recordkeeping, and for reading personnel dosimetry devices at appropriate frequencies.

Planned exposures of emergency response personnel who may receive exposures greater than 10 CFR 20 limits are on a voluntary basis and must be authorized by the Emergency Manager (or equivalent) with the advice of the radiation protection personnel.

The emergency plan provides for personnel decontamination facilities and identifies the action levels requiring decontamination. The station supplies clothing and decontamination materials to onsite personnel requiring relocation and found to be contaminated. The plan also provides an onsite contamination control program.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.12 Medical and Public Health Support

Planning Standard

> Arrangements are made for medical services for contaminated injured individuals.

Evaluation

The emergency plan describes the organizations that will provide emergency transportation, emergency medical consultation, and emergency medical services. Letters of agreement exist with all these organizations.

The plan provides for the use of the John Warner Hospital in Clinton, Illinois (approximately 6 road miles from the site), as the local hospital to provide medical treatment to emergency workers who may also be contaminated with radioactive materials or have received an overexposure to radiation.

The plan also provides for the use of the Northwestern Memorial Hospital in Chicago, Illinois (affiliated with the applicant's medical consultant - Radiation Management Corporation), as the backup hospital for more definitive care and diagnosis.

Transportation of injured individuals will be performed under agreements made with the Clinton Ambulance Service for 24-hour emergency transportation. Six local medical practitioners have also been trained in treating radiation victims, and the staff of the ambulance squad and the emergency room of John Warner Hospital will be trained by the applicant's medical consultant. The applicant will also support the local medical service personnel by providing special emergency equipment and supply kits for use for contaminated injured individuals and by providing health physics personnel to participate with and support the local medical service personnel.

The plan describes initial first-aid procedures and initial decontamination procedures that would be used on an injured worker during an emergency. Decontamination and first-aid facilities exist on site in the service building. Additional decontamination facilities exist in the control building and the radwaste building.

At least one person on each operating shift is required to have first-aid training.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.13 Recovery and Reentry Planning and Postaccident Operations

Planning Standard

General plans for recovery and reentry are developed.

Evaluation

The emergency plan contains general plans and procedures for reentry and recovery operations. The means are described by which decisions to relax protective measures will be reached by the Recovery Manager.

The Emergency Manager (in the EOF) will assess the need for initiating the Recovery Organization, once he has established the fact that the emergency conditions in and around the station have subsided. The decision to enter the recovery phase of operations will be coordinated with appropriate Federal, State, and local emergency response organizations.

Provisions exist to inform emergency workers, including Federal, State, and local authorities, that the Recovery Organization is to be initiated. Recovery operations will be coordinated in the EOF. Notification will be made to all concerned agencies whenever the Recovery Organization replaces the Emergency Organization.

The plan provides a diagram of the Recovery Organization along with the primary duties of managers in the organization. Criteria for determining when re-entry of the facility would be appropriate are identified in the plan, and a method of periodically estimating total population exposure is established using the radiation monitoring system computer.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.14 Exercises and Drill

Planning Standard

Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

Evaluation

The emergency plan provides for the conduct of periodic exercises and drills to develop and maintain emergency response skills among the various emergency response group and individual response personnel.

Annual exercises will be conducted according to the guidance set forth in NRC and FEMA rules, to test the integrated capabilities and a major portion of the basic elements within the plan. Offsite organizations as well as the applicant's response organizations will be involved. At least once every 6 years, an exercise will be started between 6:00 p.m. and midnight and another between midnight and 6:00 a.m. The scenarios used for the various exercises will contain the essential elements set forth in NUREG-0654 and and will be designed to allow flexibility in decisionmaking. Provisions exist for the conduct of unannounced exercises.

At the conclusion of each exercise, a critique will be held as soon as possible. Organizational means for evaluating the results of the postexercise critique and implementing corrective actions are established and described in the plan (see Section 5.4.5.2 of the applicant's emergency plan).

In addition to the exercise, various drills will be conducted covering communications, fires, medical emergencies, health physics, and radiological monitoring. Drills will consist of supervised instruction periods aimed at testing, developing, and maintaining skills in emergency response task areas. Management controls are established so that necessary corrective actions are implemented.

Each drill and exercise will be conducted to test the state of emergency preparedness and will be designed to meet a list of specific objectives. The Supervisor - Emergency Planning will coordinate and implement revisions to the emergency plan and required corrective actions resulting from the drills and exercises.

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.2.15 Radiological Emergency Response Training

Planning Standard

> Radiological emergency response training is provided to those who may be called on to assist in an emergency.

Evaluation

The emergency plan provides general information regarding the training of all personnel in the emergency tasks for which they are responsible. The Vice President - Nuclear is responsible for ensuring the training of all plant emergency response personnel.

The applicant will provide training and annual retraining for all emergency workers, including members of offsite organizations whose services may be required in an emergency, such as fire, police, medical support, and rescue personnel. The training will be consistent with the organizations' emergency functions.

Selected station personnel on each shift will attend the multimedia National Red Cross First Aid Course or an equivalent course.

The training program for members of the applicant's Emergency Organization will include written examinations (with minimum passing scores), practical demonstrations by each trainee of skills acquired during the training, and practice drills (as described previously in this supplement).

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

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

Planning Standard

> Responsibilities for plan development and review and for distribution of emergency plans are established, and planners are properly trained.

Evaluation

The Vice President - Nuclear has the overall authority and responsibility for radiological emergency planning. The Director - Nuclear Support is responsible for the development of the emergency response program. The Supervisor - Emergency Planning will update and improve the emergency plan and its implementation procedures as needed.

Periodic revisions of the plan (as needed), including changes identified by drills and exercise, are provided for. Provisions are also made for the distribution of plans and approved changes to all organizations and appropriate individuals.

The plan describes a continuing program for those persons responsible for the emergency planning effort to enable them to attain and maintain a state-of-the-art knowledge in the field of emergency preparedness planning. Emergency Plan Implementing Procedures are listed in Appendix B to the plan and a table of contents is provided, together with a cross-reference to these criteria (NUREG-0654). Quarterly updating of telephone numbers is provided for.

The plan provides for an independent audit of the plan and its associated Emergency Preparedness Program at least once a year in accordance with 10 CFR 50.54(t).

Finding

The staff finds that the applicant's emergency plan meets this Planning Standard; the requirements of 10 CFR 50, Appendix E; and the guidance criteria of NUREG-0654.

13.3.3 Conclusions

13.3.3.1 Applicant's Onsite Emergency Plan

On the basis of its review of the applicant's emergency plan against the Planning Standards of 10 CFR 50.47(b), the requirements of 10 CFR 50, Appendix E, and the guidance criteria in "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654, Revision 1, November 1980 (RG 1.101, Rev. 2), the staff concludes that the Clinton Power Station Emergency Plan provides an adequate planning basis for an acceptable state of emergency preparedness and meets the requirements of 10 CFR 50 and Appendix E thereto.

The criteria contained in the above documents supersedes previous Commission guidance for the upgrading of emergency preparedness at nuclear facilities. Since TMI-2 Action Plan Item III.A.1.1, related to short-term emergency preparedness, has been superseded, this issue is considered resolved.

In addition, since the applicant's onsite emergency plan satisfies the criteria contained in the above documents, the staff also concludes that the requirements of TMI-2 Action Plan Item III.A.2 are satisfactorily resolved.

13.3.3.2 Offsite Emergency Plans

The radiological emergency response plans of the State and local governments within the plume exposure pathway emergency planning zone have not yet been reviewed and evaluated by FEMA, as those plans are still under development. The Clinton site-specific Annex (Local Plan) to the Illinois Plan for Radiological Accidents (State Plan) is expected to be submitted to FEMA for review during the first quarter of 1985. FEMA's interim findings on the adequacy of the State and local plans for Clinton are expected during the second quarter of 1985. After reviewing FEMA's interim findings and determinations, the staff will provide its overall conclusions on the adequacy of the emergency planning program for the Clinton Power Station and its related emergency planning zones and will report its conclusions in a future supplement to this SER.

13.6 Operating and Maintenance Procedures

13.6.3 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

In the SER the staff reported that the applicant would implement a program of emergency operating procedures (EOPs) based on the Boiling Water Reactors Owners Group (BWROG) Emergency Procedure Guidelines (EPGs) and that the staff had developed draft guidelines for long-term upgrading of EOPs (NUREG-0799) in accordance with the TMI-2 Task Action Plan Item I.C.9. The staff guidelines were issued for public comment, the comments were resolved, and the staff issued NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," in August 1982.

On December 17, 1982, the staff modified the schedule and review requirements for the TMI-2 Task Action Plan by issuing Supplement 1 to NUREG-0737, "Requirements for Emergency Response Capability" (Generic Letter No. 82-33). Supplement 1 to NUREG-0737 required that each applicant and licensee submit a Procedures Generation Package (PGP) at least three months prior to the beginning of formal operator training on the upgraded procedures. Staff guidance for PGPs was provided in NUREG-0899. In accordance with the generic letter, the PGP must include:

(1) plant-specific technical guidelines
(2) a writer's guide
(3) a description of the program for validation/verification of EOPs
(4) a description of the program for training on the upgraded EOPs

The applicant submitted the Clinton PGP for staff comment in a letter dated May 1, 1984. The PGP included:

(1) Draft CPS Procedure No. 1450.00, Rev. 0, "CPS Emergency Procedure Guidelines," as the plant-specific technical guidelines

(2) CPS Procedure No. 1005.01, Rev. 11, "Preparation, Review, and Approval of Station Procedures and Documents," as the Clinton writer's guide

(3) Draft, "Clinton Power Station Emergency Offnormal Procedures Verification and Validation Program"

(4) "Description of the Clinton Power Operator Training Program"

In response to staff comments on the PGP materials, the applicant also provided supplemental information regarding the PGP in letters dated October 24, 1984, and January 9, 1985. The staff reviewed these materials to determine the adequacy of the applicant's program for preparing and implementing EOPs. The objectives of NUREG-0899 and the requirements of Supplement 1 to NUREG-0737 were used as the basis for the review.

13.6.3.1 Plant-Specific Technical Guidelines

The review of the Clinton emergency procedure technical guidelines materials was conducted to determine the technical adequacy of the guidelines. The review was performed by conducting a step-by-step comparison of the EPGs with the NRC-approved BWROG EPGs, Revision 3. The staff's safety evaluation report on Revision 3 was enclosed in a letter dated November 23, 1983 (D. M. Crutchfield to T. Dente). The review of the Clinton EPGs resulted in the applicant committing to make several changes in the guidelines. These resolved the staff's concerns with two exceptions. First, although the applicant agreed to revise the EPGs, no specific schedule was included or discussed in the submittals. In response, the applicant did commit in the letter of January 9, 1985, to completing a final set of EPGs to support operator training before fuel load. The staff will confirm that:

(1) the necessary revisions are made, (2) the EOPs are upgraded, and (3) operators are trained before fuel load. Second, the applicant provided, in the letter of October 24, 1984, the technical basis for the plant-specific deviation from the generic calculational method for primary containment venting pressure. The staff will review these materials and confirm their acceptability in a future supplement to the SER.

13.6.3.2 Writer's Guide

The staff reviewed the applicant's submittals addressing the writer's guide for EOPs to determine if they provided acceptable methods to meet the objectives of NUREG-0899. The staff concluded that a few objectives were not addressed in the writer's guide. In letters dated October 24, 1984, and January 9, 1985, the applicant identified the objectives and committed to revise the writer's guide to address these. The applicant provided a schedule for revision that supports completion of operator training before fuel load. The applicant is also considering including a sample EOP to address many of the formatting details of an acceptable writer's guide. The staff will confirm that the writer's guide is revised as required in time to support upgraded EOPs for operator training before fuel load. If appropriately used, the revised Clinton writer's guide should result in EOPs that are adequately usable, accurate, complete, readable, convenient to use, and acceptable to operators.

13.6.3.3 Program for the Verification and Validation of the EOPs

The Clinton verification and validation (V&V) program was reviewed using the objectives of NUREG-0899. The program consists of table-top reviews, including step-by-step comparisons of the EOPs with the EPGs, walkthroughs of the control room, and walkthroughs/talkthroughs on both mosaic mockups and the simulator. The EOP V&V program walkthroughs/talkthroughs will be integrated with the Detailed Control Room Design Review. The EOP V&V will be guided by the program's checklists and observation guides to ensure the EOPs conform to the generic guidelines and trained operators can use the EOPs to mitigate an emergency condition. On the basis of the review of the program description, it appears that the applicant's program will adequately meet the applicable objectives of NUREG-0899 and will ensure that the EOPs meet the objectives of the writer's guide. Therefore, the staff found the program acceptable.

13.6.3.4 Program for Training on the Upgraded EOPs

The applicant's description of the EOP training program was also reviewed using the objectives in NUREG-0899. The training consists of classroom instruction and will be expanded in 1985 to include simulator exercises of the EOPs. The training program includes training on the different operating philosophy of the EOPs when compared with conventional event-oriented procedures and discussions of bases for each caution and operator action. The training plan should provide operators an understanding of the EOPs, their bases, and their use. Therefore, the training plan was found acceptable.

13.6.3.5 Conclusions

On the basis of its review, the staff concludes that, with the exceptions noted in Sections 13.6.3.1 and 13.6.3.2 of this supplement concerning the technical guidelines and writer's guide, the applicant's PGP for Clinton complies with the requirements of Supplement 1 to NUREG-0737 and provides acceptable methods for accomplishing the objectives of NUREG-0899. The applicant has committed to revise the PGP, as described above. The staff will confirm this and complete the necessary review before fuel load. Future changes to the PGP should be reviewed in accordance with 10 CFR 50.59.

The staff will not review the applicant's procedures in accordance with TMI-2 Action Plan Item I.C.8, "Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants." This review is not necessary because the applicant has prepared a PGP in accordance with Supplement 1 to NUREG-0737 and the guidance contained in NUREG-0899, as discussed above. Therefore, the staff considers TMI-2 Action Plan Item I.C.8 resolved.

APPENDIX A

CONTINUATION OF CHRONOLOGY

March 2, 1984	Letter from applicant responding to Generic Letter 83-28 regarding generic implication of Salem anticipated transients without scram (ATWS). The applicant is using methodologies developed by industry groups. The schedule for information submittal will be provided within 45 days after receipt of Owners Group reports.
March 9, 1984	Letter from applicant forwarding results of review of operating shift experience.
March 12, 1984	Letter from applicant responding to Staff Questions 210.05 and 210.06 regarding heating, ventilation, and air conditioning (HVAC) duct work. The stress limits for duct work and duct supports will be included in the Final Safety Analysis Report (FSAR). Buckled portions of ducts will be replaced.
March 12, 1984	Letter from applicant forwarding "Updated Information for Antitrust Review of OL Application" per Items B.1 and B.2 of Regulatory Guide 9.3, in response to staff request of January 12, 1984.
March 23, 1984	Letter from applicant forwarding the applicant's 1983 annual financial report, the 1982 annual financial report of Soyland Power Cooperative, Inc., and the 1983 annual financial report of Western Illinois Power Cooperative, Inc.
March 30, 1984	Letter from applicant forwarding Amendment 29 to FSAR. Changes consist of revisions to fuel assembly design, fire protection, quality assurance (QA) organization responsibilities, and TMI Action Plan responses, including high-pressure core-spray (HPCS) logic modifications.
April 2, 1984	Generic Letter 84-05 issued to all power reactor licensees and applicants for operating licenses (OLs) regarding change to NUREG-1021, "Operator Licensing Examiner Standards."
April 4, 1984	Generic Letter 84-08 issued to all licensees of operating reactors, applicants for OLs, and holders of construction permits (CPs) regarding interim procedures for NRC management of plant-specific backfitting.
April 13, 1984	Summary of April 3, 1984, meeting with the applicant in Bethesda, Md., regarding emergency preparedness.

April 17, 1984	Letter from applicant advising that H. R. Victor was named Manager of Nuclear Station Engineering, effective February 27, 1984.
April 19, 1984	Generic Letter 84-11 issued to all licensees of operating reactors, applicants for OLs, and holders of CPs for boiling-water reactors (BWRs) regarding inspections of BWR stainless steel piping.
April 19, 1984	Letter from applicant providing status and schedule for resolution of preliminary design assessment human engineering deficiencies determined in November 1981 control room design review/audit.
April 26, 1984	Generic Letter 84-10 issued to all applicants for OLs regarding administration of operating tests prior to initial criticality (10 CFR 55.25).
April 30, 1984	Generic Letter 84-12 issued to all operating reactors and applicants for OLs regarding compliance with 10 CFR 61 and implementation of Radiological Effluent Technical Specifications (RETS) and attendant process control program (PCP).
April 30, 1984	Letter from applicant forwarding summary update of electrical and mechanical seismic qualification of equipment.
May 1, 1984	Letter to applicant forwarding emergency operating procedures generation package, Revision 0 to draft Procedure 1450.00, "Emergency Procedure...," and Revision 11 to Procedure 1005.01, "Preparation...," per Supplement 1 of NUREG-0737 (Generic Letter 82-33) and SER confirmatory issue 41.
May 2, 1984	Letter from applicant submitting quarterly update regarding construction schedule. Target date for Unit 1 fuel load remains January 3, 1986; commercial operation is scheduled for November 1, 1986.
May 3, 1984	Generic Letter 84-13 issued to all power reactor licensees [except Systematic Evaluation Program (SEP) licensees] and all applicants for OLs to operate power reactors regarding Technical Specifications for snubbers.
May 4, 1984	Letter from applicant forwarding "Preoperational Environmental Radiological Monitoring Program, 1983."
May 8, 1984	Generic Letter 84-09 issued to all licensees of operating reactors regarding recombiner capability requirements of 10 CFR 50.44(c)(3)(ii).
May 11, 1984	Generic Letter 84-14 issued to all operating power reactor licensees regarding requalification training program.

May 14, 1984	Letter to applicant informing that Systematic Assessment of Licensee Peformance (SALP) report findings will be discussed at May 31, 1984, meeting in DeWitt, Ill.
May 15, 1984	Summary of April 27, 1984, meeting with the applicant, GE, and Morrison-Knudsen Company regarding HPCS diesel generator test program requirements.
May 15, 1984	Letter to applicant forwarding NRC report on setpoint methodology for GE-supplied protection system instrumentation, resulting from July 14, 1983, and January 31, 1984, meetings with the Licensing Review Group and GE in Bethesda, Md.
May 22, 1984	Summary of April 18, 1984, meeting with the applicant, Sargent & Lundy, and Impell Corporation in Bethesda, Md., regarding technical approach and status of concerns regarding pool dynamics.
May 23, 1984	Letter to applicant forwarding SALP Report 50-461/84-03 for October 1982 - February 1984 for discussion at May 31, 1984, meeting.
May 23, 1984	Letter from applicant providing schedule for submittal of information in response to Generic Letter 83-28 regarding Salem ATWS issue, per the applicant's March 2, 1984, letter. The BWR Owners Group report is essentially complete.
May 25, 1984	Letter from applicant forwarding responses to SER confirmatory issue 71 (Humphrey concerns); Action Plans 5, 6, 8, and 21; and revised responses to Action Plans 2 and 3.
May 31, 1984	Letter from applicant providing further confirmation that the independent design review is consistent with the design description in FSAR, SER, and supplements. Summary of engineering design control and surveillance is enclosed.
June 1, 1984	Letter from applicant forwarding drawings for Attachment 2 to independent design review which were inadvertently omitted from the May 31, 1984, submittal.
June 5, 1984	Letter to applicant forwarding SER Supplement 3 (SSER 3) (NUREG-0853).
June 13, 1984	Letter from applicant advising that 8 full-loop tests to demonstrate step loading capability and 61 additional HPCS diesel generator tests to demonstrate steady state of performance will be performed as result of April 27, 1984, meeting in Bethesda, Md.
June 18, 1984	Letter to applicant forwarding request for additional information regarding Mark III containment design, including structural capacity of drywell and steel head to withstand positive and negative pressure differentials.

June 19, 1984	Letter from applicant forwarding protocol for use in conducting independent design review as a supplement to the May 31, 1984, review description.
June 22, 1984	Letter to applicant transmitting comments on May 31, 1984, proposed independent design review program.
June 25, 1984	Letter from applicant advising that preliminary Inspection Report 50-461/84-03 regarding SALP was reviewed. Efforts to raise management responsiveness to more acceptable level will continue.
June 27, 1984	Generic Letter 84-16 issued to all licensees of operating reactors, applicants for OLs, and holders of CPs regarding adequacy of on-shift operating experience for near-term OL applicants.
July 2, 1984	Generic Letter 84-15 issued to all licensees of operating reactors, applicants for OLs, and holders of CPs regarding proposed staff actions to improve and maintain diesel generator reliability.
July 3, 1984	Generic Letter 84-17 issued to all power reactor licensees, applicants for OLs, nuclear steam supply system (NSSS) vendors, reactor vendors, and architect-engineers (AEs) regarding annual meeting to discuss recent developments on operator training, qualifications, and exams.
July 5, 1984	Letter from applicant forwarding Amendment 30 to FSAR. Changes consist of update of radwaste process diagrams in Table 1.7-2 and Chapter 11, revision of the radwaste building HVAC system description, and revision of Appendix D regarding SER confirmatory issues 28, 38, 39, and 48.
July 6, 1984	Generic Letter 84-18 issued to all nonpower reactor licensees regarding filing of applications for licenses and amendments.
July 9, 1984	Letter from applicant advising that responses to Staff Questions 220.61, 220.62, and 220.63 will be provided by October 1, 1984, per the staff's June 18, 1984, request for information regarding Mark III containment structural capacity.
July 13, 1984	Letter to applicant advising that no changes are required to enclosed appendix to SALP Report 50-461/84-03 and acknowledging the statement regarding the applicant's intent to raise management responsiveness to a more acceptable level.
July 17, 1984	Letter from applicant requesting that the NRC review and approve independent design review program, per June 28, 1984, meeting. Bechtel is expected to complete incorporation of changes and resubmit revisions by July 20, 1984.

July 30, 1984	Letter from applicant providing quarterly update regarding construction schedule. An analysis of the critical path schedule shows that the limiting system schedule for reactor water cleanup is approximately 53 days behind schedule.
August 1, 1984	Summary of June 28, 1984, meeting with applicant, Bechtel, Schiff Hardin, Sargent & Lundy, Newman & Holtzinger, and State of Illinois in Bethesda, Md., regarding the proposed Independent Design Verification Program (IDVP).
August 6, 1984	Generic Letter 84-19 issued to all licensees of operating reactors, applicants for OLs, and holders of CPs regarding availability of Supplement 1 to NUREG-0933, "Prioritization of Generic Safety Issues."
August 6, 1984	Letter from applicant forwarding "Generic 300 Start Tests for Electro-Motive Division of General Motors (EMD) Model EMD-645E4 Diesels," and "Production Test," to substantiate HPCS diesel generator reliability per the applicant's June 13, 1984, letter.
August 6, 1984	Letter to applicant forwarding staff comments on the IDVP plan. The plan is acceptable pending resolution of enclosed comments. A meeting is scheduled for late August on the State of Illinois audit proposal regarding Contention II in licensing hearings.
August 8, 1984	Letter from applicant forwarding response to staff's request for additional information during May 22, 1984, meeting regarding SER outstanding issue 9 and confirmatory issues 13 and 71 regarding suppression pool hydrodynamics.
August 17, 1984	Letter to applicant forwarding request for additional information regarding safety parameter display system. Response is requested within 60 days to allow adequate time for onsite audits before plant licensing.
August 20, 1984	Generic Letter 84-20 issued to all licensees of operating reactors and applicants for OLs regarding scheduling for submittals of reloads that involve unreviewed safety questions.
August 22, 1984	Letter from applicant forwarding Bechtel response to NRC comments regarding Revision 1 to IDVP. The applicant concurs with the Bechtel suggestion that the response be considered as an immediately effective amendment to the plan.
August 22, 1984	Letter from applicant transmitting application to amend CPPR-137, extending latest completion date for construction of the facility to October 1, 1986, per 10 CFR 50.55(b). Series of stop work orders and recovery programs initiated to implement corrective action constitutes basis for request.

September 10, 1984	Letter from applicant forwarding "Interim Guidelines for Containment Purge Operation Clinton Power Station - Unit 1." Information in the enclosure and Licensing Review Group-II Generic Position Papers 4-CSB and 5-CSB constitute the response to SER confirmatory issue 69.
September 10, 1984	Letter to applicant responding to August 22, 1984, letter from Bechtel regarding the IDVP. Revision 1 to the program is acceptable for providing additional assurance that the facility meets licensing requirements.
September 11, 1984	Letter to applicant requesting additional information regarding TMI Action Plan Item II.K.3.28. Response is requested within 45 days.
September 14, 1984	Letter to applicant requesting additional information regarding CLASIX-3 code used to support licensing activities associated with Mark III plants as part of continuing review of hydrogen control for Mark III containments during postulated degraded core accidents.
September 18, 1984	Letter from applicant advising that F. A. Spangenberg became Director of Nuclear Licensing, effective September 13, 1984.
September 19, 1984	Letter from applicant forwarding Bechtel's September 18, 1984, proposal regarding administrative improvements in reporting aspects of the IDVP.
September 26, 1984	Letter from applicant updating seismic qualification of equipment status, per SSER 2. Schedule projections indicate 85% of equipment will be qualified and installed by May 1985.
September 26, 1984	Letter from applicant providing updated information regarding seismic qualification of equipment, per SSER 2. Approximately 56% of the equipment is qualified and 17.9% is installed. Improvements in cable pulling and termination progress should result in having 85% qualified by July 1985.
September 27, 1984	Letter from applicant confirming the position regarding State of Illinois September 5, 1984, comments on the IDVP.
September 27, 1984	Letter from applicant forwarding response to June 18, 1984, request for additional information regarding the ultimate capability of Mark III containment design (SER confirmatory issue 6).
September 28, 1984	Letter from applicant forwarding revised markup of draft Technical Specifications. Changes resulted from review of plant-specific design information in the FSAR, requirements/commitments in SSER 3 (NUREG-0853), and Technical Specifications of recent licensees.

September 28, 1984	Letter from applicant forwarding results of preliminary design assessment of control room and detailed control room design review program plan, per Supplement 1 to NUREG-0737, in response to SER confirmatory issue 38.
September 28, 1984	Letter from applicant forwarding suggested review items regarding general design control submitted with May 31, 1984, letter proposing independent design review.
October 1, 1984	Letter from applicant forwarding response to Generic Letter 83-28 regarding Salem ATWS events, per May 23, 1984, commitment.
October 2, 1984	Letter from applicant submitting Amendment 31 to FSAR.
October 2, 1984	Letter from applicant responding to August 17, 1984, request for additional information regarding safety parameter display system.
October 4, 1984	Letter to applicant confirming October 16, 1984, meeting in Rosemont, Ill., to discuss first interim report on the Bechtel independent design review, per October 2, 1984, telephone conversation.
October 9, 1984	Letter from applicant informing the staff that P. J. Telhorst, Resident Licensing Coordinator, has been assigned to work with NRC staff on questions regarding current and developing licensing issues in Washington, D.C.
October 10, 1984	Letter from applicant forwarding interim closure report regarding compliance with proposed upgrades to reactor vessel water level measurement system design. Report forms the basis for resolving TMI Action Plan Item II.F.2. Final report will be submitted by late October 1984.
October 11, 1984	Letter from applicant notifying of the availability of Sargent & Lundy analysis of independent reviews of design activities. Reviews performed by INPO, Teledyne, NRC, and CYGNA.
October 15, 1984	Letter from applicant forwarding list of design subcontractors who have assisted Sargent & Lundy in the design effort, per Item 2b of the applicant's September 27, 1984, letter.
October 19, 1984	Letter from applicant discussing October 1, 1984, telephone conversation regarding erosion monitoring efforts at construction site. Berm erosion is minor and no corrective action is planned.
October 19, 1984	Letter to applicant confirming NRC plan to perform independent verification construction inspection during November 5-16, 1984, using NRC NDE (nondestructive examination) van and contractor technicians, as discussed with the applicant on October 12, 1984.

October 22, 1984	Letter from applicant forwarding proprietary HPCS preoperational test procedure and test results for the diesel generator at Kuosheng. The Kuosheng and Clinton generators are identical.
October 24, 1984	Letter from applicant responding to staff's October 10, 1984, telephone request for information on SER confirmatory issue 71 regarding the local encroachments issue.
October 24, 1984	Letter from applicant forwarding response to the staff's September 7, 1984, comments regarding emergency operating procedures generation package, per Generic Letter 82-33 and SER confirmatory issue 41. GE is reviewing procedures per NUREG-0737, Item I.C.7.
October 26, 1984	Generic Letter 84-23 issued to all Boiling Water Reactor Licensees of operating reactors (except LaCrosse, Big Rock Point, Humboldt Bay, and Dresden 1) regarding reactor vessel water level in BWRs.
October 26, 1984	Letter from applicant providing quarterly construction schedule update. An analysis of the critical path schedule indicates that the nuclear boiler is 54 days behind schedule. Commercial operation has been rescheduled to begin on July 10, 1986.
October 29, 1984	Letter from applicant requesting November 20, 1984, meeting with NRC staff to discuss the applicability of code boundary jurisdiction between ASME Code Section III, Subsection NF, 1974 Edition, Summer 1974 Addenda, versus American Institute of Steel Construction Standards, per Inspection Report 50-461/83-09.
October 29, 1984	Letter from applicant transmitting application for an amendment to CPPR-137, deleting Section 3E(3) regarding thermal discharge regulatory limits.
November 1, 1984	Letter from applicant requesting concurrence in interpretation of paragraph on protocol regarding IDVP. Addition of paragraph regarding inclusion of NRC and Attorney General of Illinois in telephone conversations between the applicant and Bechtel is requested.
November 6, 1984	Letter to applicant confirming November 13, 1984, meeting in Rosemont, Ill., to discuss Bechtel's second interim IDVP report.
November 6, 1984	Letter to applicant forwarding Independent Design Review Implementation Inspection Report 50-461/84-39. Areas needing improvement are identified. Plans for resolution of items at November 13, 1984, meeting are requested.
November 8, 1984	Letter to applicant forwarding errata to "Modified Paragraph 6" of November 6, 1984, letter regarding inspection report.

November 9, 1984	Letter from applicant informing that Hydrogen Control Owners Group is scheduled to submit a generic plan providing additional information on CLASIX-3 code to NRC by December 21, 1984. The applicant will review plan and submit further response by January 19, 1985, per September 14, 1984, request.
November 12, 1984	Letter from applicant forwarding response to comments regarding Bechtel's first progress report on independent design review.
November 14, 1984	Letter from applicant forwarding tracking charts regarding percent of construction complete, applicant profile, nuclear power program key events, and system turnover schedule. Charts are updated through period ending September 30, 1984; quarterly updates are to be sent.
November 14, 1984	Letter to applicant forwarding request for additional information regarding TMI Action Plan Item II.D.1. Response is requested within 45 days.
November 14, 1984	Letter to applicant forwarding interim report, "Conformance to Regulatory Guide 1.97, Clinton Power Station Unit 1," based on review of applicant's September 9, 1983, submittal. Response to unjustified exceptions identified in report is requested within 45 days.
November 15, 1984	Letter from applicant discussing program to resolve local encroachment issue regarding Humphrey concerns, per SER confirmatory issue 71. Details of hydraulic control unit floor analysis and evaluations of piping and other structures are to be submitted by December 1, 1984.
November 16, 1984	Summary of November 9, 1984, meeting with the applicant and assistant attorney general of State of Illinois in Chicago, regarding Contention III concerning control room design and instrumentation for postaccident monitoring.
November 19, 1984	Letter from applicant forwarding additional information regarding TMI Action Plan Item II.K.3.28 (SER confirmatory issue 47).
November 28, 1984	Letter to applicant forwarding guidance for providing information to support NRC audit of seismic/dynamic qualification of Category I equipment and pump and valve operability qualification.
November 29, 1984	Letter to applicant advising that QA audit of Bechtel need not be performed because the November 6, 1984, Inspection Report 50-461/84-39 documents the implementation and review of the QA program and because of the possibility of jeopardizing the independence of the review.

December 4, 1984	Letter from applicant requesting revision to environmental monitoring language in Subsection 3E(1) of CPPR-137, per November 7 and 8, 1984, telphone conversations with the staff.
December 5, 1984	Letter from applicant forwarding "Reactor Pressure Vessel Water Level Measurement System Evaluation Rept" in response to Generic Letter 84-23, "Reactor Vessel Water Level Instrumentation in BWRs."
December 11, 1984	Letter from applicant forwarding Revision 1 to "Compliance Report - Regulatory Guide 1.97 (Rev. 3)," per Supplement 1 to NUREG-0737.
December 13, 1984	Summary of December 5, 1984, meeting at site regarding schedule for developing Technical Specifications.
December 13, 1984	Letter from applicant forwarding Revision 4 to radiological emergency plan and draft "Evacuation Time Estimates for Clinton Power Station Plume Exposure Emergency Planning Zone," dated July 1984. Enclosures resolve SER outstanding issue 18.
December 13, 1984	Letter to applicant forwarding lists of items resulting from review of SER (NUREG-0853) through Supplement 3, including items requiring confirmation of implementation, application, and commitments, and items potentially requiring licensing conditions.
December 21, 1984	Letter from applicant responding to SPDS preimplementation audit. Results of the action plan will be provided.
December 21, 1984	Letter from applicant forwarding description of two test configurations to be tested in 1/10-scale Mark III encroachment test facility, draft Technical Specification regarding Test Series 6104 encroachment test, and milestone schedule for SER confirmatory issue 71.
December 26, 1984	Summary of December 6, 1984, meeting with State of Illinois Attorney General in Glen Ellyn regarding forthcoming construction appraisal team inspection efforts at facility regarding Contention II (QA).
December 27, 1984	Generic Letter 84-24 issued to all licensees of operating reactors and applicants for OL regarding certification of compliance to 10 CFR 50.49.
December 28, 1984	Letter from applicant forwarding Revision 2 to emergency response capability implementation plan schedule, in response to Generic Letter 82-33. First emergency preparedness exercise date is accelerated approximately 3 months to September 4, 1985.

January 9, 1985	Letter from applicant regarding Emergency Operating Procedures Generation package (confirmatory issue 41).
January 14, 1985	Letter from applicant regarding location of high range gamma monitors.
February 1, 1985	Letter from applicant regarding automatic depressurization system actuation logic (confirmatory issue 28).

APPENDIX B

REFERENCES

Advisory Committee on Reactor Safeguards

ACRS Report No. 0938, "Report on Susquehanna Steam Electric Station Units 1 and 2," August 11, 1981.

Crutchfield, D. O.

Letter to T. Dente, Chairman, BWR Owners Group, November 23, 1983, Subject: Safety Evaluation of Emergency Procedure Guidelines, Revision 3.

General Electric Co.

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APPENDIX D

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13. ABSTRACT (200 words or less)

Supplement No. 4 to the Safety Evaluation Report on the application filed by Illinois Power Company, Soyland Power Cooperative, Inc., and Western Illinois Power Cooperative, Inc., as applicants and owners, for a license to operate the Clinton Power Station, Unit No. 1, has been prepared by the Office of Nuclear Reactor Regulation of the U. S. Nuclear Regulatory Commission. The facility is located in Harp Township, DeWitt County, Illinois. This supplement reports the status of items that have been resolved by the staff since Supplement No. 3 was issued.

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