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Division of Reactor Inspection and Safeguards

Emergency Operating Procedures Inspection

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NRC Licensee No.: NPF-43

Licensee: The Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

Inspection At: Fermi 2 Nuclear Power Plant

Inspection Conducted: July 5 through July 14, 1988

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1 INSPECTION OBJECTIVES

The inspection team reviewed the licensee's emergency operating procedures (EOPs), operator training and plant systems to accomplish the following objectives of NRC Temporary Instruction 2515/92 (Reference B.4.21)[1]:

(1) Determine whether the EOPs conformed to the vendor generic guidelines and were technically correct for the Fermi 2 nuclear power plant.

(2) Assess whether the EOPs could be carried out in the plant under the expected environmental conditions with the most limiting operating crew complement.

(3) Evaluate whether the plant staff was adequately trained to perform the EOP functions in the time available.

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[1]References and documents reviewed are tabulated in Appendix B to this report.

2 BACKGROUND

Following the Three Mile Island accident, the Office of Nuclear Reactor Regulation developed the Three Mile Island Action Plan (NUREG-0660 and NUREG-0737), which required licensees of operating plants to reanalyze transients and accidents and to upgrade EOPs (Item I.C.1). The plan also required the NRC staff to develop a long-term plan that integrated and expanded efforts in the writing, reviewing, and monitoring of plant procedures (Item I.C.9). NUREG-0899, Guidelines for the Preparation of Emergency Operating Procedures, represents the NRC staff's long-term program for upgrading EOPs, and described the use of a procedure generation package to prepare upgraded EOPs.

The licensees formed four vendor owners' groups corresponding to the major reactor vendor types in the United States: Westinghouse, General Electric, Babcock & Wilcox, and Combustion Engineering. Working with the vendor company and the NRC, these owners' groups developed generic procedures that set forth the desired accident mitigation strategy. For General Electric plants, the generic guidelines are referred to as the Boiling Water Reactor (BWR) Owners' Group Emergency Procedure Guidelines. These were to be used by licensee's in developing their procedure generation packages. The NRC has issued generic safety evaluation reports for approval of Revisions 2 and 3 of the BWR Emergency Procedure Guidelines. Revision 4 of the BWR Emergency Procedure Guidelines is now under review by the NRC. Generic Letter 82-33, Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability, directed each licensee to submit to the NRC a procedure generation package to include:

(1) Plant-specific technical guidelines

(2) A plant-specific writer's guide

(3) A description of the program for the verification and validation of the upgraded EOPs, and

(4) A description of the training program for the upgraded EOPs

Plant specific EOPs were to have been developed that would provide the operator with directions to mitigate the consequences of a broad range of accidents and multiple equipment failures. These procedures were required to be developed in a manner which considered human factors aspects.

The NRC selected a representative sample of each of the four vendor types for EOP review by teams from Regions I, II, III and IV. The NRC Office of Nuclear Reactor Regulation (NRR) selected 13 additional plants with GE BWR Mark I containments for EOP team reviews. This inspection at the Fermi 2 nuclear plant was one of the supplemental reviews conducted by NRR.

3 EMERGENCY OPERATING PROCEDURE PROGRAM EVALUATION

The inspection team reviewed the licensee's program for upgrading EOPs under Section 7 of NUREG 0737 (Supplement 1), "Upgrade Emergency Operating Procedures," to determine whether the intent of NUREG requirements was accomplished and whether the licensee submitted proper documentation to the NRC for review.

The NRC amended the Fermi 2 license (Condition 17) to require completion of selected NUREG-0737, Supplement No. 1, items. Two requirements were specified for EOPs:

(1) Prior to July 31, 1986, the licensee was required to submit a procedures generation package meeting the requirements of Section 7 to Supplement 1 to NUREG-0737 for NRC review and approval.

(2) Prior to startup following the first refueling outage, the licensee was required to complete EOP training, and have implemented EOPs based on the procedures generation package.

3.1 Procedure Generation Package Review

The licensee has committed to implement Revision 4 of the Emergency Procedure Guidelines, but had not submitted the plant specific technical guidelines for NRC review as requested by NUREG 0737 (Supplement 1).

The existing Fermi 2 EOPs were developed in accordance with an early version (Revision 1A) of the BWR Emergency Procedures Guidelines. The NRC staff reviewed these EOPs before licensing Fermi 2. In 1983, the BWR Owners' Group Procedures Committee made significant changes for the drywell spray limit, suppression pool heat load, and maximum containment pressure limit (Revision 3). The licensee revised existing Fermi 2 EOPs to include several of these changes at that time.

However, the licensee had not implemented upgraded EOPs addressed in their license condition as of this inspection. Fermi 2 does not plan its first refueling outage until 1989. The licensee provided a submittal (Reference B.4.18) to address License Condition 17. This submittal forwarded a draft Revision 0 of the Fermi 2 procedures generation package. Plant-specific technical guidelines were not submitted. Detroit Edison committed to implement Revision 4 of the Emergency Procedures Guide (NEDO-31331, Reference B.4.19), which was then still under Owners' Group review. They further committed to submit a document identifying the differences between the generic emergency planning guidelines and the Fermi 2 plant-specific technical guidelines within 3 months of NRC's approval of Revision 4.

The licensee prepared and trained operators to EOPs which conformed to NEDO-31331. The licensee also has revised the procedure generation package several times since the draft revision which was submitted to the NRC. The team was informed that implementation of the upgraded EOPs was imminent. Therefore, the team inspected the draft EOPs and the associated documents which were used to develop these EOPs. The licensee stated that their submittal (following NRC approval of the Revision 4 Guidelines) would include the actual procedures generation package used to develop the Fermi 2 EOPs.

3.2 ECP Validation Program

The inspection team performed an independent verification of the Writer's Guide development and implementation, EOP-hardware interface and EOP calculations to determine whether the licensee had properly accomplished the validation process.

Licensee validation of the draft Fermi 2 EOPs inspected by the team was conducted from April 23 through May 1, 1987. The validation was conducted under procedure POM 21.000.25, Validation Program for Emergency Operating Procedures. The Validation Program evaluated the EOPs for useability and operational correctness to verify that the procedures provided clear and correct direction to the operator to mitigate emergency events. This evaluation also confirmed that human factors considerations were effectively applied.

Evaluation for useability was intended to demonstrate that the EOPs provided sufficient and understandable information without providing superfluous information so that the operator could follow the procedures without confusion, delays, or errors. Evaluation for operational correctness was intended to demonstrate that EOP language and level of information were compatible with plant capabilities, control room and plant hardware, and minimum shift manpower levels, regardless of the emergency event.

To accomplish these tasks the licensee developed twenty-four accident scenarios to validate all steps in the EOPs. The licensee intended that an operations shift crew would respond to the scenarios on the Fermi 2 simulator using the EOPs. The scenario responses were reviewed by a validation team consisting of personnel experienced in plant operations, human factors, and engineering. The team identified discrepancies during the validation that were dispositioned in accordance with the validation program procedure POM 21.000.25, Form 2.

Because of simulator limitations, mainly in the area of primary containment simulation, eight of the twenty-four scenarios were entirely either walked-through or desk-top reviewed, as opposed to the planned intent to validate all procedures using the simulator.

The team had several observations concerning this review:

(1) The validation resulted in discrepancies that required procedural step reverifications following EOP changes. The changes were incorporated but the reverifications were never documented.

(2) Several changes to the EOPs since the original validation in 1987, although individually minor, collectively could warrant augmented validation before implementation.

(3) The licensee's validation team concluded that the text-based EOPs were an improvement but that EOP flowcharts would significantly enhance the useability of the procedures. The NRC inspection team concurred with this assessment, notwithstanding excellent performance by a shift crew during the simulator demonstration of the EOPs.

Overall, the NRC team considered that the licensee has performed a comprehensive validation of the draft procedures. The licensee identified and dispositioned validation discrepancies, including recommendations for reverifications or revalidations as necessary. The team verified that discrepancy dispositions were incorporated into the current revisions.

3.2.1 Plant-Specific Technical Guidelines versus Emergency Procedure Guidelines

The Fermi 2 Plant Specific Technical Guidelines were developed from the BWR Owners' Group Emergency Procedures Guidelines, Revision 4. A line by line comparison of the documents to identify differences requiring justification was performed by the team. Any differences were checked using the licensee's implementation items list to ensure proper justification.

The team determined there were very few unidentified differences. All except one required analyses to justify the change from the owners' group guidelines. The team reviewed these justifications and found them acceptable.

The exception involved the use of technical specification allowable values versus actual setpoint values for EOP entry conditions. The team considered this difference significant, although the actual changes in values were small. For example, the high drywell pressure trip setpoint was less than 1.68 psig; the corresponding allowable value used as an entry condition was less than 1.88 psig. The team was concerned that a valid condition could arise resulting in a protective action (e.g., small steam leak with subsequent high pressure) and the operators would not consider that the plant condition should be controlled using the EOPs because the entry condition may not be reached. Allowable values were also used throughout the EOPs as reference operating points, for example, ECCS initiation setpoints. The use of allowable values in these situations could prevent the resetting of reactor trips, isolations, and system initiations as intended by the owners' group. After discussing this difference with Fermi 2 personnel, they stated that actual setpoints would be used instead of allowable values, where conservative, before implementation of the EOPs.

3.2.2 Plant-Specific Technical Guidelines Versus EOPs

Following development of the plant-specific technical guidelines, the EOPs were developed using this guidance. The team compared the Plant Specific Technical Guidelines, POM 21.000.21 (SR) Revision 1, to the emergency operating procedures. The EOPs were accurately developed from the guidelines and the identified deviations adequately justified. However, the team found two deviations that were not documented as Fermi 2 implementation items.

(1) Plant-specific technical guidelines, Step RC/P-5 versus EOP 29.000.01, Step RC/P-5

The EOP contained an "or" statement allowing the nuclear shift supervisor to determine an alternate cooldown method. The plant-specific technical guideline did not address this option.

(2) Plant-specific technical guidelines, Step PC/H-6 versus EOP 29.000.02, Step PC/H-6.1

The plant-specific technical guidelines required a check of torus water level before starting torus sprays. EOP Step PC/H6.1 initiated sprays without a torus water level check.

The EOP validation program identified this discrepancy in April 1987. The associated form used to document discrepancies concluded that the low spray flow for Fermi 2, coupled with the fact that installed instrumentation would not measure torus water level near the spray nozzles, justified the deviation.

Both these differences appeared to be adequately resolved to the team. However the implementation lists did not identify these deviations as differences.

3.3 Verification of EOPs

The licensee's verification process was delineated in procedures 21.000.19, Procedures Generation Package (Reference B.1.5), and 21.000.24, Verification Program for Emergency Operating Procedures (Reference B.1.6). The process required a review of the plant specific technical guidelines and the emergency operating procedures. This review included compliance to the writers guide and a determination of technical adequacy. The same process was to be used, as necessary, to reverify changes.

A four-man General Electric team performed the initial verification of the procedures in March of 1987. Reverification of selected changes was designated following initial validation. These changes included addressing operator comments during simulator training, technical specification changes, plant changes, and calculation revisions.

To evaluate this process the NRC team reviewed the validation effort associated with selected procedure sections for primary containment control and secondary containment control. These sections included the source documentation for secondary containment entry conditions and secondary containment interlock defeats.

The NRC team considered that the graphs used in the primary containment control procedure required reverification following recent calculation revisions. In addition, the entry conditions for area radiation levels, HVAC exhaust radiation levels and area water levels of the secondary containment control procedure required revision to reference the appropriate pages.

The team also identified several general findings associated with the verification review. Administrative procedure 21.000.24 was not prescriptive enough to either capture references to the source documentation used in verifying entry conditions or to document the actual instruments taken credit for in the technical adequacy review. As a consequence, the licensee did not have a source document list for review to determine whether an EOP is affected when a setpoint or plant modification occurs. Reverification activities did not document a Writers Guide adequacy review.

The team found that, notwithstanding the noted omissions, initial verification of the EOPs in March 1987 was satisfactorily accomplished.

The team considered that additional guidance should be provided regarding potential sources of radiation, temperature and sump levels associated with different plant areas. In addition a sketch of reactor building areas and levels feeding each sump would be beneficial for implementation of the secondary containment control EOP.

3.4 Licensee Review of IE Information Notices

In August 1986, the NRC issued Inspection and Enforcement Information Notice 86-64, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures." This notice alerted licensees of the problems found during NRC reviews of procedure generation packages and EOPs. Information Notice 86-64 (Supplement 1), issued on April 20, 1987, described further EOP and procedure generation package problems.

The team noted that the licensee had received the supplemental Information Notice and distributed it to the appropriate organizations (Reference B.4.20). For example, the training manager was familiar with the notice which was being tracked as Operations Open Item 87-023-01. The operations engineer responsible for answering the open item was familiar with the identified issues and stated that they had been actively considered during the development of the EOPs. This engineer was also responsible for development of the upgraded EOPs.

4 PROCEDURE EVALUATIONS AND WALKDOWNS

4.1 EOP Control Room Demonstration

The team evaluated the operational useability of the EOPs during the simulator walk-throughs and demonstrations. Some EOP attributes could not be observed during the licensee demonstration because of simulator limitations, but for the most part the EOPs were shown to be workable.

In addition to these simulator reviews, a control room walk-through was performed for EOP NPP-29.000.02, Revision 5, Primary Containment Control (PC). This EOP was selected by the team because the simulator was limited in modeling the primary containment parameters during the simulator demonstration. The team performed the walk-through with a Detroit Edison simulator instructor and received support from the operations shift personnel. The entry conditions and steps of the following sections of the EOP were validated:

° Drywell Temperature Control

° Primary Containment Pressure Control

° Torus Water Level Control

° Primary Containment Hydrogen and Oxygen Control

In addition, the team reviewed selected primary containment pressure control procedure enclosures for correctness and ease of use during emergency operations. In general, the team determined that EOP NPP-29.000.02 could be used to mitigate events that would cause entry into this procedure. However, the timeliness in performing critical calculations, the effectiveness of certain actions, and incomplete directions could hamper operator responses. Examples are provided below.

(1) The first step in the Drywell Temperature Control Section of the EOP stated: "Monitor and control drywell average temperature below 145°F using available drywell cooling." The primary containment enclosure at this step was a three page calculation that can require 15-20 minutes to perform. This calculation must be repeated several times throughout the procedure. The calculated temperature was the basis for critical operator actions. The operating crew did not perform this calculation to determine drywell temperatures in the actual simulator demonstration. They simulated availability of the emergency response information system display to obtain the calculated temperature. The team considered this method to be an enhancement that should be used to obtain drywell temperature, leaving the manual calculation as a backup method.

(2) The Primary Containment Pressure Control Section of the EOP required actions based on curves of torus pressure and level (containment level) Instrumentation for the torus level did not span the full height of the torus, so calculations were necessary to determine level during certain emergency conditions. There was no explanation of how to calculate torus levels the first time this curve was referenced in the EOP. Two pages later, the procedure referred the operators to the Primary Containment Water Level Determination enclosure. This enclosure directed the operators to perform data collection and calculations to determine primary

containment water level. The calculation was not simple and required a significant time to perform.

(3) The Torus Water Temperature Control (TW/L) section of EOP NPP-29.000.02 required actions based on the average torus temperature. This was performed by summing eight torus temperature readings and determining the average value. This was a simpler calculation than that required for determining the drywell average temperature. However, the calculation was still time consuming and could divert operators from responding to the event.

(4) During the step-by-step procedure review, procedure action steps were walked-through to verify that the steps could be accomplished. The team noted that certain action steps were actually verifications. For instance, one of the first override statements for Drywell Temperature Control (DW/T) stated in part; "initiate emergency essential cooling water, isolate emergency essential cooling water to the drywell, and shutdown recirculation pumps." This step seemed to indicate that these three steps required three separate manipulations of controls by an operator. In reality, isolation of emergency essential cooling water to the drywell was an automatic action that only required confirmation by the operator. Unless training emphasized that this isolation did not require a separate action, the override could be confusing to the EOP user.

(5) The first two steps, DW/T and DW/T-1 of the Drywell Temperature Control (DW/T) section in the EOP directed the operator to control drywell temperatures. The actions required in both steps were the same, that is, start additional drywell coolers. These steps were redundant and could be combined to simplify the procedure or additional actions specified to increase drywell cooling.

(6) The Torus Water Level Control (TW/L) section of EOP NPP-29.000.02 contained a curve for the maximum primary containment water level limit. A comparison table to be used with the curve was placed below this curve. The table was too small to read, although sufficient room was available for a legible curve and table. The operator may have to locate a larger curve and table in the procedure enclosure to accomplish this step. The operator must be familiar with the organization of the procedure to know that this curve could be found elsewhere. (Full scale curves were generically placed in the procedure enclosures). Even if the user was familiar with the procedure, this step would require departure from the main steps of the procedure.

(7) Steps within the Primary Containment Hydrogen and Oxygen Control (PC/H) and Primary Containment Pressure Control (PC/P) sections of EOP NPP-29.000.2 directed the operator to use Operating Procedure 23.406, Primary Containment Nitrogen Inerting and Purge System, to vent the drywell and the torus. The team reviewed Procedure 23.406 and found that the procedure contained instructions for venting. However, it was not a simple task to locate these steps. This observation also applied to the useability of Procedure POM 23.415, Drywell Cooling System. The team considered that the licensee should review the interface between the EOPs and the operating procedures to ensure clarity.

(8) Throughout the inspection the licensee stated that the primary containment wetwell would be called the "torus" rather than the "suppression chamber or pool" terminology used by the owners' group. The Fermi 2 Implementation Items List stated that "torus" would be used instead of "suppression chamber or pool" because this is the correct terminology for a Mark I containment. However, during the walk-through the team identified that a number of controls and indicators in the control room used the terminology "suppression pool". For example, entry condition alarms, temperature recorders, the emergency response information system, and the narrow range pressure recorders all used the terminology "suppression pool".

The team also noted that several other indicators and controllers used the terminology suppression pool. This difference in terminology, from a human factors standpoint, could lead to confusion during the performance of the EOPs during an actual event.

4.2 Walkdown Validation of EOPs

The inspection team conducted walkdowns of selected portions of the EOP support procedures which directed activities outside the control room. Appendix B.1 of this report includes a listing of the procedures verified during the walkdowns. The walkdowns were conducted with licensed operators who were knowledgeable of EOP requirements based on recent training. The operators who would actually be expected to perform the procedures were not used since they had not yet received training on these activities. The team examined the following attributes.

- Adequacy of procedure guidance
- Ability of the operators to perform the procedures
- Availability of special tools and equipment in the plant
- Material condition of the systems and equipment being operated by the procedures

Additionally, the team walked down the various pathways for venting the containment as described in the support procedure.

4.3 Technical Adequacy of Procedures

The overall technical adequacy provided by the EOP enclosures appeared to be adequate. For those procedures that were already verified, plant operators were able to walk-through the tasks for the inspection team. Several of the procedures were in the process of being validated for useability and availability of equipment. An independent assessment of the capability of performing these procedures was difficult. The team did identify the following deficiencies with the procedures during the walkdowns:

Procedure 29.000.01 Alternate Boron Injection, Section 2 - Boron Injection with the Standby Feedwater System

The team walked down this procedure in conjunction with the EOP validation conducted by a licensed operator. Numerous minor deficiencies were noted by the operator and the team. These deficiencies were identified for correction before final issuance of the procedure. The more significant deficiencies that were noted are discussed below.

(1) One of the prestaged tool boxes in the turbine building was too tall for the operator to easily lift equipment out of it. The licensee committed to reduce the vertical height of the box to correct this problem.

(2) The procedure required the layout of about 400 feet of non-collapsible hose. The hose connected the precoat tank, located on grade level of the turbine building, to the suction of the standby feedwater pumps, located on the lower level of the turbine building. The hose was made up in 50 foot sections using screw-type couplings. Some of the couplings would not freely rotate, making it difficult to couple the sections together. Additionally, the licensee made no provision to vent air from the line before starting the portable pump and opening the suction valve to the standby feedwater pump. This approach could lead to air binding of the pump. The licensee committed to add a step to backfill the hose with water and to free the couplings to enable quick connection of the hose sections.

(3) The operator could not loosen the cap on the suction line to Standby Feedwater Pump B with reasonable force applied by a pipe wrench. The licensee issued a work request and corrected the problem.

(4) The operator experienced difficulty moving pre-staged drums of borax and boric acid into the plant from the warehouse. Drums and equipment blocked the pathway which was marked for emergency access. The licensee took corrective action to clear the pathway. The licensee committed to conduct surveillance of this pathway by the Turbine Building Auxiliary Operator to be sure clear access was maintained.

(5) The team observed that the pallets of borax and boric acid could not be placed immediately adjacent to the pre-coat tank because of their weight and the presence of railing around the operating platform. The team was concerned that carrying multiple buckets of granulated chemical to the tank via the steps from the floor to the platform would be very tiring and time consuming and could result in gross spillage of the chemical. The licensee concurred and committed to provide means via forklift extenders to place chemical barrels on the platform adjacent to the tank.

Procedure 29.000.01 Alternate Boron Injection, Section 3 - Boron Injection with Condensate/Feedwater Systems

The team walked down this procedure with a licensed operator conducting EOP validation. The operator and the inspection team noted minor deficiencies for correction before final issue of the procedure. The operator discussed a potentially significant deficiency concerning the use of the cold feed injection flow path through Valve N21-F606 if the reactor vessel pressure was less than 200 psig. Completion of this step may not be possible due to the valve being inoperable under these conditions. The valve is normally closed and de-energized, and had failed to operate as required on a previous occasion. The licensee committed to revise the procedure to add steps which re-energize valve controls and minimize the differential pressure across the valve before opening. The preventive maintenance program requires stroking the valve on a refueling outage frequency.

Procedure 29.000.01 Alternate Control Rod Insertion

A portion of this procedure directed the venting of control rod drive over piston volumes in order to insert control rods that were not fully inserted upon scram. The team observed the vent ports would be connected with thin wall tygon tubing and routed to a radioactive waste drain. The team was concerned that this venting operation could result in the discharge of high temperature water or steam in the area occupied by the operator if the tygon tubing became incapable of containing the effluent because of heat and pressure. The licensee committed to study this concern and provide a resolution.

Procedure 29.000.01 Alternate Boron Injection Section 1 - Boron Injection from Reactor Water Cleanup Using the Standby Liquid Control Tank

The procedure directed the operator to run a 250 foot line from the standby liquid control pump relief valve return line to the reactor water cleanup pre-coat tank. The team noted that the point of connection of the line to the standby liquid control system was elevated about 3 feet with respect to the bottom of the standby liquid control tank. The lower portion of the tank must be siphoned in order to deliver the full amount of boron for cold shutdown conditions (23" tank level per emergency procedure guidelines Appendix C calculations). The licensee justified the acceptability of this routing by stating that the pump would run continuously, rather than intermittently as indicated in an earlier version of the procedure. A valve on the temporary discharge line would control flow into the pre-coat tank.

The inspection team observed that the procedures requiring implementation outside the control room did not note that the procedures assumed normal power supplies to be available in order to conduct the necessary operations. For example, in Sections 1 and 2 of the Alternate Boron Injection procedures, the portable pumps were to be plugged into the nearest 120 volt ac power outlet. The procedures requiring implementation inside the reactor building did not note that radiation levels may preclude or limit entrance to the building. The licensee concluded that inability to implement reactor building steps would not have a significant impact on accident mitigation activities, since other means of controlling torus water level, injecting boron, inserting control rods or compensating for loss of water level instrumentation were provided. The inspection team concurred with this conclusion.

4.4 Availability of Special Tools and Equipment

The availability of tools and equipment in the plant was difficult to assess because of the pre-implementation validation activities being conducted concurrently with the inspection. The team noted that special EOP boxes were prepared and placed in the control room and plant areas to contain equipment and procedures. Pumps, hoses, and couplings were available. Proper connections could be made. Drums of borax and boric acid were pre-staged on pallets in a nearby warehouse. The drums were available for delivery by forklift to the pre-coat tank area. During the walk-downs, the team observed that tools such as screwdrivers, pipe wrenches, hammers, knives, and scoops for boric acid were generally not pre-staged at the time the walkdowns were conducted. Inventory sheets for tools and equipment were not in place, preventive maintenance instructions and shelf life considerations for equipment were not instituted and procedures did not reference locations of the EOP boxes. The licensee committed to develop inventory sheets and perform periodic surveillance, revise procedures to state locations of EOP boxes, incorporate preventive maintenance and shelf life considerations and pre-stage necessary tools.

One area of significant strength was the implementation of procedures for interlock defeats. The licensee had established a separate folder for each interlock defeat procedure. The folder contained the procedure, all required panel or cabinet keys to access fuses or terminal strips listed in the procedure, and marked jumpers for each jumper to be installed by the procedure. These were contained in the Control Room EOP box, along with screwdrivers and electrical tape for installing jumpers or lifting and insulating leads. Additionally, the licensee had marked the points at which terminal changes were to be made. Red tags mounted inside the cabinet stated the action required and referenced the applicable procedure.

4.5 Plant Material Condition

The inspection team reviewed the material condition of the plant during tours and walkdowns to be sure that necessary equipment and components were accessible and functional. The general condition of the turbine, auxiliary, and control building areas visited was acceptable. Reactor building areas below the fifth floor were still contaminated from a primary system leak that occurred several weeks before the inspection. Thus, anti-contamination clothing was required to enter these areas. In addition, numerous areas had leaks that were collected via collectors and tubing to plant drains. Access to the reactor water level transmitters was restricted due to contamination restrictions. The team observed less than optimum housekeeping and post-work cleanup practices beyond the expected housekeeping problems produced by the contamination cleanup. For example, tools and equipment were generally left scattered.

The noise levels in the reactor building were generally high. These levels could be significantly higher following an accident when emergency equipment would be operating. Communications could be conducted via the plant public address system or portable radios, but effective communication could be difficult in a post-accident environment. The licensee told the team that the plant public address system was to be improved. Planned additions included multiple channels and more call stations, along with sound isolation booths in selected areas.

The identification and tagging of major valves and components, excluding instrumentation and associated valving, was a strength. Tags were well-attached, large and easily readable, and contained adequate information. In addition, area location maps were present so people could readily determine equipment position relative to building areas and floor elevations.

4.6 Verification of Calculations and Setpoints

The team reviewed the calculation package for figures and setpoints used in the EOPs based on the Owners' Group calculational procedures. The team also reviewed selected Detroit Edison Company design calculation packages for instrumentation setpoints used as entry conditions to the EOPs. These packages included the setpoint determination for the fuel pool exhaust ventilation isolation, the evaluation of water sources for long term recirculation cooling following a loss of coolant accident (Regulatory Guide 1.82), and the determination of primary containment water level in the enclosure to Procedures 29.000.01 and 29.000.02. The results of the reviews are summarized below:

(1) The calculational package for figures and setpoints used in the EOPs was generated during June 1988 using the latest version of Appendix C to the emergency procedure guidelines. The licensee supplied the input values to General Electric, who performed the calculations, verified the input to the calculational programs and the resulting curves, setpoints and numbers. The licensee appears to have obtained current, proper, and validated curves and setpoints for use in the EOPs with the following exception. The team noted that the torus and drywell parameters for the torus vent paths and the drywell vent paths were reversed. The licensee's verification process did not correct this discrepancy. During the inspection the licensee confirmed that General Electric used the correct values in their calculations and obtained documentation to support the conclusion.

(2) The team reviewed design calculation packages that validated surveillance procedures for drywell pressure, reactor dome pressure, and reactor water level to determine if the technical specification values for Nominal Trip Setpoint and Allowable Value were conservative for the instrumentation channels installed in Fermi 2. The design calculations were based on and consistent with the method presented to the NRC by the BWR Owners' Group in NEDO-31336. The calculations provided a rigorous justification of setpoint values, setpoint tolerances, error determination, and avoidance of spurious trips. The calculations also considered harsh environment effects on instrument accuracy. About 75 of these calculations had been performed. The team considered the extent and detail of these calculations a strength.

(3) The team reviewed the determination of the setpoint value of the fuel pool ventilation exhaust radiation monitoring system trip setpoint and the instrumentation surveillance procedure for the trip channels. The value calculated by Radiological Engineering was 6.1 mrem/hr. The determination stated that the trip setpoints must be no greater than that value. The team noted that the surveillance procedure allowed setpoints to be as high as 6.25 mrem/hr. The licensee committed to revise the surveillance procedure to a maximum trip setpoint valve of 6.1 mrem/hr, consistent with the setpoint determination.

(4) The team reviewed the licensee's evaluation of the potential for loose insulation and other debris to clog the suction of emergency core cooling system pumps from the suppression pool (Regulatory Guide 1.82). The licensee prepared and submitted this evaluation in response to NRC question 042.10 during the licensing process. The evaluation concluded that any insulation entering the suppression pool, either metal or metal-encased fibrous material, would tend to sink to the bottom of the suppression pool below the suction strainers of the emergency core cooling system pumps. Additionally, the basket-type strainers were designed to pass adequate emergency core cooling system water at design net positive suction head conditions while 50% plugged. Flow velocities were not expected to result in pinning of debris to the strainers. The basket shape does not permit a single piece of debris to restrict flow to more than one side or area of the strainers. The licensee did not show calculated flow velocities and impact on debris, nor was the amount of fibrous material expected to be

discharged to the suppression pool during a loss of coolant accident quantified. However, the team considered that the conclusions were reasonable and that net positive suction head conditions would not likely be worse than those considered in emergency core cooling system design and in the EOPs.

The licensee had considered the possibility of debris generation in the drywell at pressures greater than 20 psig and torus water temperatures greater than 120 degrees Fahrenheit. The EOPs require tripping drywell cooling fans when conditions exceed these values due to unqualified coatings (paint) on the drywell fan housing interior.

(5) Procedures 29.C00.01 and 29.000.02 contained an EOP support procedure for Primary Containment Water Level Determination in case the reactor pressure vessel level cannot be determined. Step RC/L-5.4 required maintaining primary containment water level between 631 feet and the Maximum Primary Containment Water Level Limit. The support procedure provided for determination of primary containment water level up to 630 feet using drywell and torus pressures from instrument T50-R802A/B and verification of level at or above 630 feet by continuous water flow out of the drywell pressure indicator test fitting. No instrumentation was available to accurately ensure water level could be maintained above 631 feet. The licensee stated that this item had been considered in the detailed control room design review.

5 SIMULATOR EXCERCISES AND TRAINING

5.1 Simulator Validation of EOPs

The inspection team validated portions of the EOPs using the licensee's site specific simulator. The licensee provided qualified licensed operators and simulator instructors to support the validation. These scenarios were designed to test the maximum number of EOP decision paths during the available simulator time. The scenarios were not suitable for testing licensed operator performance. Event sequences were aggravated by using malfunctions beyond the design bases of the plant.

The inspection team members were given a simulator familiarization demonstration on the afternoon of July 7, 1988. On July 8, 1988, the team observed a shift of six control room personnel performing three scenarios written by the inspection team. An extra operator assigned to the shift was rotated out of the scenarios to stay within minimum shift complement.

The objectives of these scenarios were to observe:

(1) the interface between the control boards, the operating staff, and the emergency procedures

(2) place keeping mechanisms used by the operating staff, and

(3) the level of expertise with which the operating staff performed the emergency procedures

Following each scenario a round table discussion was held with the inspection team, the simulator instructors, and the operating crew. The following paragraphs describe the scenarios.

5.1.1 Scenario One

Scenario one started with the plant at 100 percent reactor power when the main steam isolation valves received an isolation signal. The control rods would not insert. The main steam isolation valves closure removed the feedwater pumps and the main condenser. Because all the energy was then sent to the torus, torus water level and temperature had to be controlled. The torus temperature and pressure increase also affected the drywell environment, as did a steam leak simulated in the drywell. All means to control torus and drywell parameters were removed by malfunctions. Injection of standby liquid control was delayed by a malfunction. This delay required the level/power control procedure to be used by the crew. Emergency depressurization was required because of the torus temperature. Reactor water level was then controlled by the low pressure injection systems.

5.1.2 Scenario Two

Scenario two was initiated from 100 percent reactor power when a steam leak was developed in the drywell. The automatic reactor trip malfunction prevented an automatic scram from the consequent drywell pressure increase. A manual reactor trip was required. The pressure increase was supposed to automatically start all four diesel generators, however diesel generator 11 failed to start. After the scram, the hydrogen concentration in primary containment increased above the value requiring venting. Attempts to use the standby gas treatment system were defeated. Increasing hydrogen concentration required emergency depressurization of the reactor vessel.

5.1.3 Scenario Three

Scenario three required entry into the radiation release emergency procedures. An uninsolable steam leak in the turbine building was designed to require emergency depressurization. Stuck control rods required the use of alternate rod control and insertion. The simulator performance prevented the need for emergency depressurization, however this aspect was discussed during a table-top review after the scenario.

5.1.4 Conclusions

The team had no concerns as a result of the observation of the three simulator scenarios. The full shift complement provided to perform the scenarios used the procedures in a facile fashion, communicated well, and maintained excellent control of plant conditions through the procedures. This shift had just completed two weeks of training on the draft EOPs.

The nuclear assistant shift supervisor was always in control. The nuclear shift supervisor operated in an "overseer" position. The team did observe that the crew had some difficulty communicating the drywell and torus hydrogen and oxygen parameters. The team also noted that the crew had some difficulty reading the drywell and torus pressures from the front panels.

5.2 EOP Training

The inspection team reviewed sixteen training scenarios used by the training staff for requalification training (tabulated in Appendix B).

The team concluded that the training of licensed operators on Revision 4 Emergency Procedure guidelines was not fully integrated and only exercised mild transients. The scenarios did exercise multiple procedures, but only to the first few steps. For example, during safety relief valve operation torus water temperature was not allowed to rise past the value requiring emergency depressurization (heat capacity temperature limit). The four anticipated transient without scram scenarios were from a maximum of 40 percent power and the main condenser always remained available. During reactor vessel water level casualties, a total loss, or significant delay, of emergency core cooling injection and feedwater never occurred (steam cooling - step RC/P-6). The main condenser was lost during only one scenario.

During interviews with the training staff the team concluded the reason for the limited dynamic training was simulator limitations. The staff wrote the training scenarios around the limitations of the simulator to prevent "negative training." The team considered this rationale acceptable.

The team's review of training schedules showed that the emergency procedures were either dynamically exercised or table-top discussed by all active licensed operators.

The team's review of training material and training schedules revealed that the power plant operators (non-licensed) had been or would be trained on the use of emergency enclosures and emergency procedures before the procedures were implemented.

6 CONTAINMENT VENTING

The team reviewed the EOP support procedure for emergency primary containment venting. The procedure was entered from Procedure NPP-29.000.02, Step PC/P-4 to control pressure below the Primary Containment Procedure Limit or from Step PC/H-5, 1.2 to control hydrogen and oxygen at elevated pressures. Direction was given to the operator to vent through the torus vent path first, if available. If the torus could not be vented, then the operator was directed to vent the drywell. The torus path was preferred because of the scrubbing effect of the torus water on radioactive gases and particulates that may be present. For pressure control, Step PC/P-4 also directed the venting operation to be conducted irrespective of offsite radioactivity release rate and to defeat interlocks if necessary to accomplish the venting operation.

The procedure for pressure control vented to maintain pressure below 54-56 psig, depending on primary containment water level. The licensee chose this limit based on the design pressure of the drywell and torus. They had not performed an analysis to determine the ultimate capability of the primary containment. In the procedure for Emergency Primary Containment Venting, the licensee chose successively larger vent paths using the hard-piped vent paths from the drywell and torus to the suction of the standby gas treatment system. This path consisted of a 20-inch diameter line off the torus and a 24-inch line off the drywell. The line reduced from 24-inches to 16-inches diameter outside the reactor building at the inlet to standby gas treatment system. Upon initiation of standby gas treatment system, valves would open automatically to take suction both from the reactor building exhaust system through a 24-inch line and from the air inlet in the 5th floor refueling area through a 20-inch line. Additional connections to this line come from the nitrogen inerting and supply system through 1-inch, 2-inch, and 6-inch lines.

To vent the torus, the procedure directed the operator to initiate or verify operation of the standby gas treatment system. If this system was not operating or operable, the operator was directed to open or verify open suction valve T46-F410 to the reactor building fifth floor air inlet. This action allowed dispersion of the discharge into the reactor building air space. To subsequently vent the torus, the 20-inch upstream torus vent valve (T46F400) was first opened, along with the 6-inch downstream torus vent bypass valve (T46F412). This action effectively opened a 6-inch vent path from the torus. If this path was insufficient to control primary containment pressure, then the operator opened the 20-inch downstream torus vent valve (T46F401). If additional venting capability was required, the operator then opened the 20-inch torus supply air purge inlet valve (T48F404) and the 20-inch torus purge inlet valve (T48F405), which vents directly to the torus room. The standby gas treatment system was capable of processing this discharge via the reactor building exhaust system connection.

Venting of the drywell was similar, in that the vent path through the 6-inch outboard drywell vent bypass valve (T48-03-F602 and T46F411) was opened first, then the 24-inch outboard drywell vent valve (T46F402), and finally the 24-inch drywell supply purge inlet valve (T48-03-F601) and the 10-inch drywell air purge inlet valve (T48-F407).

During the review of the procedure and associated drawings for the standby gas treatment system and the nitrogen inerting system, the team noted the following concerns:

(1) The vent paths chosen for the procedure were not all the available vent paths from the primary containment. The licensee did not consider several 1-inch, 2-inch and 6-inch vent paths to the standby gas treatment system. These paths could be effective in initially reducing pressure under some accident scenarios where pressure increases were not rapid. Use of these paths would not cause pressure concerns in other areas of the vent path, particularly the standby gas treatment system.

(2) The procedure cautioned that the venting evolution may release radioactive gas or steam into the reactor building. When the 6-inch path and particularly when the 20-inch and 24-inch paths vent directly to standby gas treatment system, pressure may be greater than the 2 psig design pressure of the standby gas treatment system. This system is located in the auxiliary building outside of secondary containment.

(3) The licensee did not perform an engineering analysis of the vent paths chosen for the procedure. The licensee could not determine the relative flows to standby gas treatment system, the reactor building fifth floor or the reactor building exhaust system ductwork and the resultant pressures in these areas. Thus, the team could not be assured that the standby gas treatment system, exhaust system ductwork, or the reactor building blowout panels would remain intact or that valves would operate under the projected differential pressures.

(4) The licensee did not consider opening the vent path from primary containment to the torus room early in the procedure. This action could allow dispersion of the pressure into the large volume of secondary containment. Subsequent processing of the effluent via the reactor building suction paths to standby gas treatment system would not cause failure of these systems.

(5) The procedure assumed availability of electrical power and control air for manipulation of valves and controls. No direction was given regarding manual operations, hookup of nitrogen or air backup supplies to open or close valves or other means to ensure venting could be accomplished under these degraded conditions.

(6) The 24-inch inboard drywell vent valve (T48-03-F602) was inoperable and had been so for several weeks. All technical specification requirements for inoperable containment isolation valves were met. There were no requirements to maintain an operable vent path from either the torus or the drywell in the current technical specifications.

The team reviewed a study conducted by a consultant to the licensee that scoped the capability of the plant to vent primary containment under conditions of a severe accident. The study addressed possible modifications to provide a full capability for venting with pressures as high as 90-100 psig. The licensee did not currently have plans to implement the recommendations of this report.

In a related issue to primary containment pressure control, the licensee installed a modification to isolated emergency essential cooling water on high drywell pressure. This isolates cooling water to the reactor recirculation pumps and the drywell coolers to limit heat input to the cooling water system to essential equipment services to design valves. The team did not question the acceptability of the modification, which required justification as a significant deviation from the Owner's Group Guidelines which direct initiation of all available containment cooling under these conditions. However, the team was concerned that the EOPs did not provide direction to re-establish cooling water to the drywell coolers in accident scenarios where emergency essential cooling water heat loads from other sources were not so large that operation of essential equipment would be jeopardized. The Owners' Group emergency procedure guidelines state to operate all available drywell cooling to maintain or reduce drywell temperature and pressure before initiation of engineered safety features. The team believed the accident response could be improved if drywell cooling could be re-established.

7 HUMAN FACTORS CONSIDERATIONS

7.1 EOP Writer's Guide Review

The team reviewed the Writer's Guide for Emergency Operating Procedures (POM 21.000.20 - Rev. 1), considering guidance provided in NUREG-0899, Guidelines for the Preparation of Emergency Operating Procedures. The review brought several concerns to light. Since revision of the Writer's Guide should not have a delaying effect on implementation of the revised EOPs, the team recommended that the concerns be considered promptly for revision of the Writer's Guide.

The concerns judged to be of immediate significance were:

(1) The Writer's Guide needed to provide expanded descriptions and examples of the proper and improper use of conditional and logic statements (NUREG-0899 Section 5.6.10 and Appendix B).

(2) The Writer's Guide did not specify that the place-keeping aids (blanks for checkoffs) should be placed along the right margin of the page on which they appear, rather than at the end of the text, where they are difficult to locate (NUREG-0899, Section 5.5.4).

(3) The Writer's Guide did not specify the criteria to be used in deciding whether steps should be referenced or included in the body of the EOP. (For example, "Include any steps from operating procedures that require less than a single page.") (NUREG-0899, Section 5.2.2).

(4) The Writer's Guide did not specify the criteria to be used in developing and presenting flowcharts as job performance aids or diagnostic aids to support the EOPs (NUREG-0899, Section 5.5.9)

(5) The EOPs reviewed by the team were of excellent legibility. However, to maintain that legibility in future revisions, the Writer's Guide should include specifications for type font and size to be used in printing EOPs.

(6) The Writer's Guide did not specify the criteria to be used in clearly specifying "Non-sequential", "Recurrent", and "Time Dependent" steps; the methods for clearly identifying to the operator the conditions under which these steps apply; the time sequences or intervals at which these steps are to be performed; nor the conditions under which these steps should no longer be carried out (NUREG-0899, Sections 5.7.2 through 5.7.8).

There were other concerns identified that may not be of major significance. However, the team considered that these concerns should be evaluated by the licensee for future revisions of the Writer's Guide. These concerns included:

(1) Section 6.4 of the Writer's Guide addressed capitalization. It is easy to confuse "initial capitalization" with capitalizing all letters of a word. Use of the phrases "All Capitals" or "Upper Case" to refer to instances when all the letters of a word are to be printed in upper case would clarify the approach.

(2) The Writer's Guide could specify that the page number, as well as the reference, for non-emergency procedures should be given. The bulk of the text in these procedures is not relevant to the EOP use of those steps.

(3) The Writer's Guide could state specific criteria for providing location for out-of-control-room components.

(4) The Writer's Guide could specify that the procedures are page numbered in a fashion that provides the total number of pages (e.g., Page 7 of 12).

7.2 Plant Walkdowns

The team performed walkdowns of the control room and balance-of-plant local control stations that would be used in the execution of the EOPs. The team identified some strengths and several weaknesses in the human engineering aspects of both control room and local control stations as result of these walkdowns. The specific strengths were:

The control room was found to be well-designed in general, from the human engineering viewpoint. It was roomy enough to avoid conflicts in movement, the lighting system was excellent, and the mimic and demarcation on the control panels appeared to be very good.

The ongoing program of labelling, location and availability of local control procedures, and providing tools, jumpers, etc. for local control actions appeared to provide good human engineering for local control actions.

The team found the following weaknesses during these walkdowns:

(1) The color bands that had been added to many meters as a result of the detailed control room design review were too narrow to be very useful. They were difficult to see from any distance and were not visible at all from the center control panel because they did not extend far enough to be seen past the corner of the vertical in-line meters.

(2) The three-pen recorder for DRYWELL PRESSURE-NARROW RANGE, DRYWELL PRESSURE-WIDE RANGE, and TORUS PRESSURE-WIDE RANGE (Instrument T50-R802 A,B) had three range readings. The top and middle scales showed a narrow range of -5 to +5 psig (0 in the center), and a wide range of 0 to 250 psig for drywell pressure. The second action pressure for drywell was 20 psig, which was less than 10% scale indication on the middle scale but off the upper scale.

(3) The two-pen recorder for DRYWELL DIFFERENTIAL PRESSURE and SUPPRESSION POOL DIFFERENTIAL PRESSURE (instrument T48-R808) had two weaknesses. The first weakness was that "suppression pool" had been discarded as a usable term in this control room and had been changed to "torus" elsewhere on the control panels. The second weakness was that this recorder was the only narrow range indicator for torus pressure in the control room and was scaled in inches of water, whereas the wide range indicator was scaled in psig.

7.3 EOP Useability

The team observed the human engineering aspects of the useability of the EOPs during evaluation of the EOPs in simulated operations. Specific strengths of EOP useability were found to be that:

(1) The "lay-flat" binding system was judged to be good

(2) The legibility of the EOPs was judged very good to excellent, and

(3) The use of miniature figures in the body of the EOPs was judged very good to excellent, with the exception noted below.

Weaknesses identified regarding the useability of the EOPs were:

° The Writer's Guide, Administrative Procedures, operator training and EOPs themselves did not provide a preferred method of place-keeping beyond checkoff blanks. The team observed operators following several sections of single procedures and several procedures concurrently. The operators used 3M "Post-it" sticky notes to mark their place. Subsequent interviews revealed that other operators use paper clips, pens stuck in the procedure books, and removal of looseleaf pages (NOT a recommended method) for place-keeping. The team considered that the licensee should evaluate and identify a preferred place-keeping method that is procedurally supported and trained on a periodic basis.

° EOP step PC/P-2 for initiation of torus sprays had a logical condition statement that had a "WHEN - BUT ONLY IF" structure. The implication of WHEN (as stated in the Writer's Guide, Subsection 4.7.2) was that the operator must hold at that point and not perform subsequent steps. If the torus was full (above 54") subsequent steps including containment venting could not (by procedure) ever occur. This weakness was exacerbated because "BUT ONLY IF" should only be used with "IF" in an "IF - BUT ONLY IF" structure. However, the Writer's Guide did not state this. At the same time, the team was told by training personnel that operators were trained to proceed past "WHEN" statements, rather than hold.

7.4 Human Factors Review of EOP Format and Production

The team reviewed the EOPs to determine the basic human factors quality of format and production. More strengths than weaknesses were found here. The formatting and production of the EOP drafts that were reviewed by the inspection team were very good. The weaknesses pertained to editing problems more than generic human factors problems. Particular strengths that were noted included:

° The override statement format was judged to be very good.

° The use of miniature figures in the body of the EOP was judged very good except for level conversion tables on some figures that were almost unreadable. These same tables on the full-sized figures in the enclosures to the EOPs were also of poor legibility.

- There was good and consistent handling of action statement continuations to succeeding pages. Cautions and notes were repeated when they applied to succeeding pages.

However; the team did note a need for a technical editor to review the documents. The team noted page number references which were not accurate; the use of "bolding" for significant words was inconsistent; bullets and asterisks to identify override statement continuations were inconsistently applied; and capitalization of such verbs Open and Close was inconsistent.

7.5 Interview Results

The team interviewed personnel that would be using, or would be affected by, the EOPs under review to obtain information about their preception of the EOPs. The team interviewed a cross-section sample of nine personnel:

JOB CLASSIFICATION	(LICENSED/UNLICENSED)	Number
Operations Engineer	(SRO)	1
Nuclear Shift Supervisor	(SRO)	1
Nuclear Assistant Shift Sup	(SRO)	1
Shift Technical Advisor	(SRO-certified)	1
Nuclear Supervising Operator	(RO)	1
Nuclear Power Plant Operator	(unlicensed)	1
Procedure Writer	(unlicensed)	1
Training Technician	(SRO-certified)	1
Health Physics Technician	(unlicensed)	1

The general areas of questioning and the consensus responses were:

- Role/Task Definition

  The roles of personnel in the control room and in using the EOPs were clearly specified in Detroit Edison Company Administrative procedures. Those interviewed uniformly understood these roles and tasks without any confusion. This response was supported by observations of crews at work in the control room and in the simulator.

- Use of EOPs

  There were no problems in using the EOPs in the control room. Laydown space was ample; location, labeling, etc., were all adequate.

- Procedure Content

  Technical adequacy was considered to be good to very good. Clarity was considered to be very good to superior by those interviewed. Transitions and place-keeping were the one weakness reported by those interviewed. They reported that place-keeping was difficult and that there was no standardized method of place-keeping in use. They also reported that this concern had been reported to the Nuclear Safety Review Group, and perceived that a resolution would be forthcoming.

- Communication

Interviewees reported the perception that communication - both within the control room and with personnel in the balance-of-plant and local control stations - was very good to superior. The perception was reported that communications would become even better once installation of the new, multi-channel paging system was completed.

- Control Room Environment

  This area was perceived to be very good to superior under all conceivable conditions.

- Training

  Those interviewed reported that they felt training on these EOPs was very good to superior and that they felt quite comfortable that they were adequately trained and that the training would continue to be very good.

- Validation and Verification

  Most of the personnel interviewed had provided comments on various aspects of these EOPs during training and reported confidence that their comments were being adequately considered. In addition, many of the those interviewed had been involved in the validation and verification efforts. The system for implementing changes into the EOPs was clearly understood by those interviewed.

- Additional Open-Ended Questions

  The team asked two additional open-ended questions: "Do you feel that you have enough time to execute the EOPs?" and "Given everything we have discussed, are you comfortable and confident that procedures will work and can be used in an actual event? - Why?" The answers to both questions were confident, even enthusiastic, affirmatives. The operations personnel felt that their concerns and questions had been adequately addressed, that their training was well-planned and executed, and that the EOPs to be implemented would be of superior quality.

APPENDIX A

MEETING ATTENDANCE

NAME	TITLE	ORGANIZATION	7/5/88*	7/14/88**
Pat Anthony	Compliance Enginneer	DECo	X	X
Gene Preston	Operations Engineer	DECo	X	
Joseph H. Plona	Ops. Support Engineer	DECo	X	X
Ralph Architzel	Sr. Ops. Eng. Team Ldr.	NRC/NRR	X	X
David B. Waters	NRC, Consultant	Prisuta-Beckman	X	X
Paul R. Farron	NRC, Consultant	NEC	X	X
Greg R. Overbeck	Dir., Nuc. Training	DECo	X	X
Paul Fessler	Dir., Plant Safety	DECo	X	
Frank Svetkovich	Technical Engineer	DECo	X	
Ralph Andersen	Rad. Protec. Manager	DECo	X	X
Walt Rogers	SRI, Fermi 2	NRC	X	X
W. M. Tuotov	Supt., Operations	DECo	X	X
Lynne Goodman	Director, Nuc. Lic.	DECo	X	X
Martin Virgilio	Project Director	NRC/NRR		X
Charles J. Haughney	Chief, RSIB	NRC/NRR		X
Donald L. Shurman	NRC, Consultant	INEL	X	X
K. Michael Spencer	NRC, Consultant	INEL	X	X
R. B. Stafford	Director, NQA	DECo		X
Douglas R. Gipson	Plant Manager-Fermi	DECo		X
B. Ralph Sylvia	Sr. Vice President	DECo	X	X
Stanley G. Catola	Vice President, Nuc. Eng. and Services	DECo	X	X
John Tibai	NSRG - Staff Engineer	DECo		X
Dedras K. Mohan	Nuclear Eng. Prin. Eng	DECo		X
J. E. Cohen	Shift Tech. Advisor	DECo		X
C. R. Gelletly	Gen. Sup.- Plant Eng.	DECo		X
Ted Quay	Project Manager	NRC/NRR		X
F. E. Abramson	Supervisor, QPA	DECo		X
Brad M. Williamson	Nuc. Training - Lead Instructor	DECo		X

* An entrance meeting was conducted which addressed the scope and plans for the inspection

** An exit meeting was conducted at the conclusion of the inspection. Team members presented their observations for each area inspected and responded to questions from licensee representatives. The licensee was informed that some of the observations could become potential enforcement findings.

APPENDIX B

DOCUMENTS REVIEWED

1 Procedures

1.1 NPP-23.406, Primary Containment Nitrogen Inerting and Purge System, Revision 21

1.2 POM 23.415, Drywell Cooling System, Revision 7

1.3 POM 21.000.20, Writers Guide for Emergency Operating Procedures, Revision 1

1.4 POM 21.000.25, Validation Program for Emergency Operating Procedures, Revision 0

1.5 POM 21.000.24, Verification Program for Emergency Operating Procedures Revision 0

1.6 NPP-21.000.19, Procedures Generation Package (PGP), Revision 1

1.7 POM 21.000.21, Plant Specific Technical Guidelines (PSTG), Revision 1

1.8 NPP-20.000.21, Reactor Scram, Revision 12

1.9 NPP-29000.01, Revision 5, RPV Control

EOP Support Procedures walked through:
Alternative Boron Injection Alternate Control Rod Insertion
Interlock Defeats
Section 1, Defeat ADS Auto Initiative
Section 6, Defeat of MSIV's and Main Stream Line Drain Valve Isolations
Section 7, Defeat of RWCU Isolations
Section 9, Defeat of RPS Logic Trips
Section 10, Defeat of HPCI Low RPV Pressure Isolations
Section 12, Defeat of RHR Shutdown Cooling Isolations

1.10 NPP-29.000.02, Revision 5, Primary Containment Control

EOP Support Procedures walked through:
Emergency Primary Containment Venting Primary Containment Water Level Determinative Interlock Defeats
Section 1, torus Water Management Interlock Defeat

1.11 NPP-29.000.03, Revision 5, Secondary Containment Control

EOP Support Procedures walked through:
Interlock Defeats
Section 2, Defeat of Turbine Building HVAC High Radiation Isolation

1.12 NPP-23.406, Revision 21, Primary Containment Nitrogen Inerting and Drug System

1.13 NPP-23.404, Revision 10, Standby Gas Treatment System

1.14 NPP-23.415, Revision 7, Drywell Cooling System

1.15 NPP-44.020.101(SQ), Revision 20, NSSS - Fuel Pool Ventilation Exhaust Radiation Monitor, Division I, Channel A Functional Test

1.16 EFO-8080, Revision 2, Operator Aids

1.17 NPP 44.010.017 (SQ), Revision 20, RPS and NSSSS - Reactor Vessel Low Water Level (Level 3), Division 1, Channel A1/A Calbiration, 3/26/88

1.18 NPP 44.010.0051 (SQ) Revision 20, RPS - Reactor Steam Dome Pressure, Division 1, Channel A1/A Calibration, 3/22/88

1.19 NPP 44.010.037 (SQ), Revision 20, RPS and NSSSS - Drywell Pressure. Division 1, Channel A1/A Calibration, 3/25/88

1.20 MI-IC-3003, Revision 0, Filling the Reactor Vessel Level Indication Reference Legs Under Emergency Operating Conditions

2 Training Documents

2.1 08-02-05-00, EOP Implementation Plan, Revision 0
2.2 08-02-05-02, EOP Cautions, Graphs, Calculations, Revision 0
2.3 08-02-05-03, EOP - RPV Control (RC), Revision 0
2.4 08-02-05-05, EOP - Secondary Containment Control and Radioactivity Release Control (RR), Revision 0

3 Emergency Operating Procedures Exercise Guides

3.1 09-05-50-11 Loss of all High Pressure Injection, Revision 1
3.2 09-05-50-12 Failure to Scram, Revision 1
3.3 09-05-50-14 PC Pressure Control - Spray, Revision 1
3.4 09-05-50-18 Level/Power Control 100% ATWS, Revision 1
3.5 09-05-50-22 Torus Water Level Falling, Revision 1
3.6 09-05-50-27 Loss of All High Pressure Injection System, Revision 0
3.7 09-05-50-28 Torus Water Level Falling, Revision 0
3.8 09-05-50-29 Secondary Containment Rad Level (SC/R), Revision 0
3.9 09-05-50-30 Loss of Feedwater (RC/L-1), Revision 0
3.10 09-05-50-31 RFPT Trip - Fail to Scram (RC/Q-1), Revision 0
3.11 09-05-50-32 Fail to Scram - SCL Injection (RC/Q-1), Revision 0
3.12 09-05-50-33 MSIV Insulation (RC/P-1), Revision 0
3.13 09-05-50-35 Fail to Scram - Stuck Open Relief Valve, Revision 0
3.14 09-05-50-36 Fail to Scram - RFPT Trip (C2-2), Revision 0
3.15 09-05-50-37 Steam Leak Drywell (PC/H-1), Revision 0
3.16 09-05-50-38 Steam Rupture - Containment, Revision 0

4 Miscellaneous Documents

4.1 Summary of EOP changes as a result of 1988 requalification training

4.2 Ltr. to R. Bournet fr. M. Hoffman dated 5-27-88, "Inplant Training for Emergency Operating Procedures", Shift 6 Cycle 4-885; Training Schedule

4.3 Fermi 2 Validation Program Completion, June 5, 1987

4.4 Human Factors Report, Fermi 2 Validation, May 20, 1987

4.5 Primary and Secondary Containment Venting Policies at Fermi 2, May 27, 1987

4.6 Letter, June 21, 1988, J H Plans from LC Fron, EOP Questions on Drywell Cooling

4.7 Safety Review, SGC 88-0068, April 11, 1988, Modification of EECW and EESW System Controls to provide an automatic initiation in the event of high drywell pressure.

4.8 UFSAR Change Notice 88-079, July 11, 1988, Regulatory Guide 1.82 Information

4.9 Drawing No. 6M721-5737, Revision E, Standby Gas Treatment System Functional Operating Sketch

4.10 Drawing No. 6M7215739-1, Revision G, Nitrogen Inerting System Functional Operating Sketch

4.11 Letter, July 13, 1988, File from LC Fron, Sources of EOP Input Data

4.12 Letter, June 11, 1985, RL Andersen from W.V. Lipton, Setpoints for Fuel Pool Ventilation Exhaust Radiation Monitoring System

4.13 Letter, June 10, 1988, Joe Plona from Jason Post, GENE, Revised EPG Revision 4 Appendix C Calculations

4.14 Letter, May 31, 1988, J. Plona from L.C. Fron, Input Data Validation - EOP Curve Basis

4.15 Letter, April 27, 1987, A. K. Lim from F. A. Lehnert, NUTECH, Severe Accident Containment Policy Scoping Evaluation for Fermi 2 - Final Report.

4.16 Design Calculation DC 4529, Volume 1, Revision B, Drywell Pressure Surveillance Procedure Validation

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4.18 Letter, July 31, 1986, E. Adenson from F. Agosti, Submittal of Procedures Generation Package and Modified License Amendment Request

4.19 NEDO-31331, March 1987, BWR Owners' Group Emergency Procedures Guidelines, Revision 4

4.20 Memorandum, April 27, 1987, J. Leman from J. Myquist, Forwarding IE Notice 86-064 Supplement 1

4.21 NRC Temporary Instruction 2515/92, Emergency Operating Procedures Team Inspections, April 5, 1988