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Division of Reactor Inspection and Safeguards

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Inspection At: Browns Ferry Nuclear Station
Athens, AL

Inspection Conducted: August 8 through August 18, 1988

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1.0 INSPECTION SCOPE

The inspection was performed to verify that the Browns Ferry Nuclear Power Station (BFN) emergency operating procedures were technically accurate; that their specified actions could be physically carried out in the plant using existing equipment, instrumentation, and controls; and that the plant staff could correctly perform the procedures. The team also reviewed the licensee's provisions for containment venting. The inspection was conducted in accordance with the guidance in Temporary Instruction (TI) 2515/92, "Emergency Operating Procedures Team Inspections."

2.0 BACKGROUND

Following the Three Mile Island (TMI) accident, the Office of Nuclear Reactor Regulation developed the "TMI Action Plan" (NUREG-0660 and NUREG-0737), which required licensees of operating plants to reanalyze transients and accidents and to upgrade their emergency operating procedures (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 describes the use of a procedures generation package (PGP) to prepare EOPs.

The licensees formed four vendor owners' groups corresponding to the four major reactor types in the United States: Westinghouse, General Electric (GE), Babcock and 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 GE plants, the generic guidelines are referred to as emergency procedure guidelines (EPGs). These EPGs were to be used by licensees in developing their PGPs. Generic Letter 82-33, "Supplement 1 to NUREG 0737 - Requirements for Emergency Response Capability," required each licensee to submit to the NRC a PGP that included:

(1) plant specific technical guidelines with justification for safety-significant differences from the EPGs

(2) a plant specific writers' guide

(3) a description of the program to be used for the verification and validation of the EOPs

(4) a description of the training program for the upgraded EOPs.

The licensees were to develop plant specific EOPs that would provide the operators with directions for mitigating the consequences of a broad range of accidents and multiple-equipment failures.

For various reasons, there were long delays in obtaining NRC approval of many of the PGPs. Nevertheless, the licensees have all implemented their EOPs. To determine the success of this implementation, a series of NRC inspections are being performed to examine the final product of the program: the EOPs. A representative sample of each of the four vendor types was selected for review by four inspection teams from Regions I, II, III, and IV.

An additional 13 facilities having General Electric Mark I containments, including BFN, were selected for EOP review. These inspections are being conducted by the Office of Nuclear Reactor Regulation and include a detailed review of the containment venting provisions of the EOPs.

3.0 DETAILED INSPECTION FINDINGS

3.1 Program and Procedure Review

The emergency operating procedures at BFN were identified as emergency operating instructions (EOIs). Documents reviewed during the inspection are listed in Attachment B.

The team evaluated the BFN program and its end product, the EOIs, with respect to the requirements of Supplement 1 to NUREG-0737, "Requirement for Emergency Response Capability," and the guidelines in NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures."

Before this inspection, the NRC Human Factors Branch and its contractor had reviewed the BFN procedures generation package (PGP). NRC's contractor, Pacific Northwest Laboratories, reviewed the PGP that had been submitted by the licensee on June 22, 1984, and prepared a technical evaluation report (TER) dated June 1986. As a followup to this review, the NRC staff and its contractor visited the BFN site. The findings from that site visit were documented in a TER that was issued on May 2, 1988. On July 20, 1988, the licensee submitted a revised PGP to the NRC for review. Attachment IV of this revised PGP addressed the findings in the TER issued on May 2, 1988. The team's review of the BFN PGP included these documents.

3.1.1 Comparison of BFN Plant-Specific Technical Guidelines With BFN EOIs

The team reviewed the approved BFN EOIs and related contingency procedures and compared them with the BFN plant-specific technical guidelines (PSTGs) for Unit 2. Deviations identified during the review included the following:

(1) Step RC/L-1 in the EOIs included the diesel generators as a system to be checked and activated. The BFN staff had specifically eliminated reference to the diesel generators in this section of the PSTGs on the basis that the injection systems at BFN did not depend on ac power, (e.g., high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems).

(2) Step RC/L-1 in the EOIs did not include the caution contained in the PSTGs to "place the applicable residual heat removal service water (RHRSW) pumps in service as soon as possible" after initiating low pressure coolant injection. This action was more important at BFN than at some other BWRs because there was no bypass line around the residual heat removal heat exchangers.

(3) Step RC/L-2.1 in the EOIs read, "Maintain level above -150 inches (+20 inches on LI-3-52 and 62A)." The same step in the PSTGs read 0 inch rather than +20 inches.

    In addition to this numerical difference between the PSTGs and the EOIs, this step, which was repeated in several places throughout the EOIs, should be revised to include the numbers for both level instruments that were referred to (i.e., LI-3-58A and B for the -150-inch reading; see Item 7 below).

(4) The table of alternate injection subsystems that accompanies step RC/L-3 of the EOIs indicated a discharge head of 150 to 0 psig for the RCIC and HPCI systems when being supplied with steam from the auxiliary boiler. The PSTGs indicated 350 to 0 psig.

(5) The PSTGs indicated a value of 931.3 psig in Step RC/P-1 as the pressure limit after safety relief valves (SRVs) were manually opened. While a value of 931.3 psig would be impossible to read on the installed instruments, a value of 930 psig would be readable. The EOIs contained the generic value of 950 psig.

(6) Several steps in the EOIs referred to the action to be taken if the control air supply to SRVs was lost. An example was Step RC/P- 2.1a which read, "If the continuous SRV air supply is or becomes unavailable, place the control switch for each SRV in the CLOSE or AUTO position." The PSTG step used the words "DW Control Air and Control Air." In addition to revising the EOIs to reflect the current PSTGs, the licensee should add an additional phrase indicating that the containment atmosphere dilution system nitrogen can now be cross-connected to drywell control air.

(7) The EOIs contained several references to the Yarway-type level instruments in the control room (2-LI-3-58A and 2-LI-3-58B). The following excerpts from the EOIs all referred to these instruments:

- Page 1 of 2 in Contingency C2, "Emergency Rx Depressurization"

```
*************************************************
*  Heated reference leg instruments [red labels]   *
*  are not reliable during rapid Rx                *
*  depressurization below 500 psig.                *
*  For these conditions, use cold reference        *
*  leg instruments [green labels] to monitor Rx    *
*  water level.                                    *
*************************************************
```

- The table on page 2 of 6 in Step RC/L referred to these instruments as

  Emergency Systems Range (-155 to + 60 in.)

- In other parts of the EOIs (e.g., Steps RC/L-2.3, C1-7.3, and C5-2a) the use of the words "...Rx water level above -150 in." assumed that the operator understood that these Yarway instruments were the ones to observe.

- The actual labels for these instruments in the Unit 2 control room were black on gray and read as follows:

2-LI-3-58A
Reactor Water Level A
ACDT RANGE

- The labels on these instruments in the simulator were black on yellow and read:

2-LI-3-58A
Reactor Water
LEVEL A

The BFN staff should coordinate the control room labeling project, maintenance of simulator fidelity activities, and nomenclature in the procedures to arrive at a consistent nomenclature and set of labels for these instruments.

(8) In Contingency C2, "Emergency Reactor Depressurization," Step C2-1.3 directed the operator to "defeat isolation interlocks if necessary, using one or more of the systems on the facing page." On the facing page (page 2 of 2), the references had the correct information, but were listed in a sequence that differed from that on the system list provided earlier.

(9) The diagram on page 4 of EOI-1, which showed the organization of all the EOIs and supporting contingency procedures, did not include the procedures in EOI-3, "Secondary Containment Control and Radioactivity Release Control."

3.1.2 Comparison of Revision 2 of BFN Writers' Guide With NUREG-0899

The team reviewed Revision 2 of the BFN writers' guide (WG). In general, the WG complied with NUREG-0899. However, the team identified a number of deviations of the BFN WG from NUREG-0899.

(1) The WG (Section 4.2.8, page 11) did not differentiate between the exclusive and the inclusive word "OR." The exclusive "OR" is equivalent to "A or B" but not both.

(2) Using the word "THEN" at the end of an action to instruct operators to perform another action within the same step runs actions together (e.g., "Open the valve THEN close the breaker"). Actions that were embedded in this manner could create several problems: embedded actions may be overlooked, may be confused with logic statements, and may be more difficult to verify with the single checkoff.

3.1.2.1 Guidelines for Logic Diagrams and Flowcharts

(1) Item 6 of Section 4.3.8, page 11, of the WG stated, "If multiple uses of logic terms are required to describe a condition or action, a logic diagram should be used rather than a conditional statement." An example of where this was not followed was on page 3 of Step RC/P-2.1.

(2) The WG did not discuss capitalization in logic diagrams. Text should be written in both capital and small letters for two reasons: (1) if all words are capitalized, then capitalization cannot be used for emphasis, and (2) text written in all capital letters was more difficult to read than one written in both capital and small letters. An example was Step RC/L-2.3.

(3) The WG did not state that notes and cautions in logic diagrams would be placed in the flow path directly before the steps to which they apply.

(4) The WG did not specify the size of type that will be used in logic diagrams.

3.1.2.2 Referencing

(1) Section 4.4.1.3, page 12, of the WG discussed referencing, but did not distinguish between a reference that directed operators to return to the original procedure and one that did not.

(2) The WG did not discuss methods, such as the use of tabs for easily identifying the sections and subsections of the EOIs.

3.1.2.3 Vocabulary and Syntax

(1) Section 4.3.3, page 9, of the WG stated that short, simple words should be used in EOIs. For this reason, the verb "commence" in Appendix A should be replaced with "begin."

(2) Only action verbs listed in Appendix A of the WG should be used. These verbs should be used with consistency in the EOIs. For example, the words "initiate" and "start" should not be used interchangeably.

3.1.2.4 Formatting of Action Steps

(1) Section 4.4.1, pages 12 and 13, of the WG did not define and provide the format of the following types of action steps: nonsequential steps, equally acceptable steps, recurrent steps, time-dependent steps, and diagnostic steps.

(2) Section 4.2, page 6, of the WG discussed major steps and substeps, but the WG did not discuss the relationship between major steps and substeps. Major steps should be used as headings that summarize the actions in the associated substeps. Only substeps should contain specific operator actions.

(3) Item 6 of Section 4.2.3, page 6, of the WG stated that both override steps and "steps containing an unexpected action" shall be enclosed in boxes. If these two types of steps were formatted identically, operators could mistake a step containing an unexpected action for an override step. Furthermore, because override steps must be remembered after they have initially been read, they should be formatted in a unique manner.

3.1.2.5 Figures and Tables

(1) The WG did not identify how figures would be labeled.

(2) The WG did not state that all figures would be reproduced from originals and would be of quality equal to that of the originals. A review of the EOIs showed that the reproduction of figures was poor and, in most cases, the figures were small, making them difficult to use especially under reduced lighting conditions.

3.1.2.6 Identification and Location

(1) Section 4.3.5, page 10, of the WG indicated that information on location should be provided, but did not address the format to be used when presenting this information.

(2) Section 4.3.5, page 10, of the WG discussed component identification codes but did not discuss the format to be used when including these codes in procedures.

3.1.2.7 Organization of EOIs

(1) Section 4.1.1, page 3, of the WG discussed the information to be included on the EOI cover sheet, but did not include a requirement for the revision number and date.

(2) Section 4.1.3, page 3, of the WG discussed entry conditions, but did not specify the format to be used when presenting entry conditions. The WG should specify this format and provide an example.

3.1.3 Comparison of EOIs With Revision 2 of BFN Writers' Guide

The team compared in detail the EOIs with the writers' guide (WG). In general, the EOIs conformed to the WG and the deviations did not seriously degrade the useability of the EOIs. The most important area in which the EOIs needed improvement was in the way logic statements were written. The team identified a number of instances where the EOIs did not follow the WG. Examples are provided below:

(1) Logic terms were used incorrectly throughout the EOIs. The terms "IF" and "WHEN" were frequently used without using "THEN." The terms "AND" and "OR" often were not emphasized correctly. Terms such as "before" and "except" were emphasized and used as logic terms.

(2) Item 6 of Section 4.3.8 of the WG stated that in those cases where multiple uses of logic terms were required, a logic diagram should be used rather than a conditional statement. Logic diagrams were used only once in the EOIs. Use of logic diagrams would improve the useability of the EOIs.

(3) The figures and graphs throughout the EOIs were difficult to use because of their small size and the poor quality of reproduction.

(4) Terms in the EOIs such as "are," "stop," and "can" were not emphasized consistently.

(5) The cautions in the EOIs were not numbered. For example, the full-page caution on page 2 of Step RC/L had no number or title.

(6) Figures in the EOIs were sometimes not titled correctly. For example, Figure D defined conditions under which emergency reactor depressurization was required. The figure was entered with two variables, pressure and water level, but was entitled "Pressure Suppression Pressure." Another example was Figure F, "Suppression Pool/Temperature."

3.1.4 Review of EOI Calculations

The Boiling Water Reactors Owners' Group emergency procedure guidelines (BWROG EPGs) included a number of plant-specific limits, setpoints, and action levels that required calculation of plant-unique values. Appendix C of the EPGs provided detailed directions for developing input data and performing these calculations.

The team reviewed a sample of the input data development and final calculation and verification documentation for the final Appendix C calculations listed below. The team verified the correlation of input data from EPG Table C1-T4, performed checking calculations, and confirmed to the extent possible that the calculations had been performed in accordance with the EPG Appendix C procedures. The following calculations were reviewed:

- low pressure coolant injection and core spray net positive suction head
- drywell spray initiation pressure limit
- pressure suppression pressure
- primary containment design pressure
- primary containment pressure limit
- drywell spray flow rate
- suppression pool cooling spray initiation pressure.

The team noted that these calculations included input assumptions, bases, and the identification of the performer. The team also found that all but four of the calculation packages had received a documented independent review. These four calculation packages had been identified by the licensee as requiring further revision and/or review for various reasons and were expected to be completed before plant restart.

The calculations had been performed, reviewed, and approved by the Nuclear Production organization, not the Division of Nuclear Engineering. Although the team considered the performance and review of the calculations technically acceptable, the licensee had no procedure available for the control of the calculation activities. This is discussed further in Section 3.1.6.

3.1.5 Ongoing Evaluation of EOIs

Section 6.2.3 of NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," recommends that licensees establish a program for the ongoing evaluation of the EOPs. The licensee established an ongoing evaluation program by implementing Plant Manager's Instruction (PMI) 12.6, "Implementation and Maintenance of EOIs." Section 4.8 of this instruction described the dynamic, ongoing EOI evaluation process. This section also delineated a method for anyone using or interfacing with the EOIs to make known their concerns or comments by filling out an EOI evaluation sheet, which was provided as Attachment 2 to the procedure.

However, interviews with licensee operations personnel indicated that they were not familiar with the program for suggesting improvements to the EOIs. None of those interviewed had filled out an EOI evaluation sheet.

3.1.6 Quality Assurance for Plant-Specific Technical Guidelines

NUREG-0899, Section 4.4, "Quality Assurance," states that the plant-specific technical guidelines (PSTGs) should be subject to examination under the BFN overall quality assurance (QA) program to ensure that they are accurate and up to date.

The existing PSTGs and associated appendices and calculations were not controlled under the licensee's QA program, although the EOIs themselves were. Site Director Standard Practice (SDSP) 2.11, "Implementation and Change of Site Procedures and Instructions," Revision 8, provided the primary controls for plant procedures and applied to the EOIs. In addition, the licensee had implemented Plant Managers Instruction (PMI) 12.6, "Implementation and Maintenance of EOIs," Revision 2, to add specific features required to ensure that PSTG revisions, verification and validation requirements, and training are implemented for EOI revisions. These procedures were generally acceptable for control of the EOIs.

However, the licensee had not implemented procedural controls for the preparation, review and approval of the procedures generation package (PGP), PSTGs and PSTG Appendices A through D. These documents had not been issued or administered as controlled documents in accordance with the TVA QA Program. The TVA Operations Support Group EOP Coordinator advised the team that current plans included creation of an "EOI Program Manual," which would include the PGP, PSTGs, and appendices and would be subject to either the controls of SDSP 2.11 and PMI 12.6 or other equivalent controls. The licensee further stated that a calculation control procedure would also be developed and applied to control the calculations of PSTG Appendix C.

3.2 Containment Venting

The team reviewed the provisions in EOI-2 for conformance with the BWROG EPGs, acceptability of the engineering bases for the procedures, and the ability of the operators to implement the procedures during walkthrough scenarios.

EOI-2, Step PC/P-1 required initial venting of the containment in accordance with EOI Appendix 13, "Venting Primary Containment," when drywell pressure reached 2.45 psig. The first vent path used was 2-inch piping from the suppression chamber followed by venting through 2-inch piping from the drywell. If venting through these paths was unsuccessful, Step PC/P-5 required emergency venting through the large-bore (20- to 24-inch) flow path in accordance with EOI Appendix 17, "Containment Venting Bypassing Interlocks," if drywell pressure exceeded 55 psig. This pressure was chosen on the basis of maintaining operability of the safety relief valves. The initial flow path selected was from the torus with a second flow path available from the drywell.

The licensee had reviewed the design of the vent pathway and standby gas treatment (SBGT) system piping and ductwork for the large-bore flow paths and had concluded that expansion joints in the high-pressure and low-pressure transition in the duct immediately downstream of the outboard containment isolation valves would fail, resulting in venting of the containment to the reactor building. The licensee viewed this as a desirable consequence in that it would prevent overpressurization damage to the SBGT system and keep it available to process the reactor building atmosphere for elevated release.

However, no specific analyses or tests had been performed to evaluate building pressurization, ground-level radioactivity release, and similar considerations.

Additionally, the licensee had not evaluated the operability of the containment vent path isolation valves under postaccident differential pressures (damper 64-36, valves FCV-64-32, 64-29, 84-19, 84-20, and others as listed in EOI Appendices 13 and 17). At the time of the inspection, the licensee was gathering the information necessary to determine whether the valve actuators, torque switch settings (as applicable), etc., could open and reclose the valves under expected containment venting flow and differential pressure conditions.

Similarly, the EOIs did not consider local/manual venting under blackout conditions or loss of offsite power. The licensee stated that no major changes in methodology or approach were planned until NRC approved Revision 4 of the BWROG EPGs.

3.3 Plant Walkdowns of EOIs

To ensure that the EOIs could be successfully implemented, the team performed walkdown evaluations of all the EOIs and supplemental procedures referenced in the EOIs. It verified that EOI instrument and control designations were consistent with the installed equipment and that indicators, annunciators, and controls referenced in the EOIs were available to the operators. The team verified that the controlled set of EOIs in the control room was easily accessible to the operators. It also verified that activities that might be required outside the control room during an accident could be performed. Because of the status of the BFN plant, tools, jumpers, and test equipment to support the implementation of the EOIs had not been staged in the plant. The licensee stated that dedicated EOI tool and equipment boxes were planned for installation before plant restart. The team also reviewed postaccident radiation and environmental considerations made by the licensee relating to local operations in the reactor building, and these are discussed in Section 3.5 of this report.

During the plant walkdowns, the team identified the following discrepancies:

(1) Provisions for remote in-plant communications were inadequate to support implementation of the EOI appendices. Most of the EOI appendices required manipulation of valves and equipment in the reactor building and auxiliary instrument rooms. The operations personnel involved in the team's EOI walkdowns stated that these operations would most likely be directed by phone or radio from the control room. For example, EOI Appendix 10, "Locally Venting Control Rod Drive Withdrawal Lines," required entry into the reactor building and into a contaminated area and involved extensive valve manipulation on a possible combination of 185 control rods.

    Discussions with plant operators indicated that the hand-held radio system was not effective for communicating in all areas of the plant. Although dial phones were located throughout the plant, their availability was not ensured during loss-of-power events. Similarly, sound-powered phone jacks were located throughout the plant, but neither handsets nor headsets appeared to be readily available and the licensee was unable to tell the inspection team whether the systems would remain operational during loss-of-power events.

The operations superintendent stated that a new in-plant radio system with repeater and hand-held units was being acquired and installed; this should address the above concerns.

(2) The EOIs and appendices did not provide any information on valve location and frequently did not include the name of the valve. The operators had difficulty in finding about 20-percent of the valves to be manually operated and required assistance from other operators or the control room. Examples included the following:

- valves 85-614 (control rod drive vent valve for each hydraulic control unit)

- valves 74-681A and B (condensate transfer to RHR)

- valves 74-622 and 74-624 (drain pump discharge to loop crosstie line), which were located above the torus.

To minimize unnecessary delays and congestion of the communications system during an accident, the licensee should consider providing information on the valve locations in the procedures or in a place that is convenient for the operators.

Seven manual valves that had to be operated in Step 1 of Appendix 7, Subsection g, were all located in areas of the core spray room that were very difficult to access. Electrically and mechanically, this alternate injection lineup was probably the most difficult to perform of those listed in Appendix 7. Either modifications should be made or this lineup should be noted as being the alternate injection option of last resort.

(3) Several of the valves required to be locally operated by EOI Appendix 7, "Alternate Injection Subsystems," were inaccessible from the reactor building floor or existing gratings. Examples included the following:

- Appendix 7, Subsection a - valves 75-582A (condensate transfer to loop I core spray), 2-713, 2-712 (condensate transfer to RHR). No prestaged ladders or reach rods, etc., were available.

- Appendix 7, Subsection h - valve 12-777 (steam supply to Unit 2 reactor core isolation cooling system was located about 20 feet above floor level. Same as above.

- Appendix 7, Subsection j - valve 12-778 (steam supply to Unit 2 high pressure coolant injection system) was located about 15 feet above floor level. Same as above.

The licensee stated that access provisions had not been evaluated as part of the procedure validation process but would be reviewed.

(4) The temperature instruments (TI-80-34-6 and TI-80-34-7 and 8) specified in the Caution 1 table (page 2 of 6, Step RC/L-1) for determining temperature in the vicinity of the various level instrument reference legs were physically located on the back panels of the control room. Both in the control room and in the simulator, operators used the temperature instruments available on the front panels to make these determinations. The licensee

should consider methods of correcting this situation, such as moving the instruments to the front panels or performing a calculation that would show conservative equivalency between the back and front-panel instruments.

As part of the detailed control room design review, the licensee has installed phenolic placards on the glass faces for the above containment temperature indicators and recorders on the rear of main control board panel 9-4. The placards listed the indicated or recorded data points but obscured most of the charts or dials, preventing the operators from observing recorded trend information without opening the instrument door. The licensee stated that the placard installations would be reevaluated.

(5) EOI-3 included a table of secondary containment area temperature instruments and the maximum normal operating values for each area (e.g., high pressure coolant injection (HPCI) room) representing an EOI entry condition. The licensee stated that the maximum normal operating values listed for each instrument corresponded to the alarm setpoints for the individual channels and had been verified and validated.

However, a sample of five of the alarm setpoints were checked via direct readout at panel 9-21, and the following four setpoints were found to exceed the EOI entry condition value in the above table:

- HPCI room, El. 519 - 185°F as found rather than the 175°F entry condition
- northwest corner room, El. 519 - 182°F as found rather than the 175°F entry condition
- southwest corner room, El. 519 - 168°F as found rather than the 160°F entry condition
- torus area, El. 519 - 180°F as found rather than the 175°F entry condition.

Since the indicators for these instruments only alarmed in the control room and were only readable, one at a time, from a common monitoring panel behind the main control board, the erroneously high setpoints could cause late recognition of the fact that the entry condition had been reached.

The licensee subsequently stated that the indicators were verified to be within the cumulative instrument channel maximum tolerances and were consistent with EPG and PSTG guidance, but that several of the setpoints had been recently changed and not yet incorporated into the EOIs. The licensee could not tell the team why the setpoints had been changed without determining the effect on the EOIs.

(6) During walkdown of EOI Appendix 12, "Alternate Depressurization/Pressure Control," the HPCI turbine steam supply mimic on the vertical portion of panel 9-3 incorrectly showed the HPCI turbine stop valve (FCV-73-18) as being downstream of the HPCI turbine control valve (FCV-73-19). The licensee stated that the mimic would be reviewed and corrected if required.

(7) The low-range drywell pressure instrument (PI-64-67B) and associated recorder (R-64-50) were scaled for absolute pressure in a range of 0-80 psia while the wide-range drywell pressure instrument (PI-64-160) was scaled for gauge pressure in a range of 0-300 psig. These instruments were variously used for determining EOI entry conditions, EOI limits, initiation

of containment venting, etc. Some EOI steps and curves (e.g., entry conditions specified in Figure C, "Drywell Spray Initiation," of EOI-2.1) only provided gauge values, others (e.g., EOI-2, SP/L, PC/P-5, Appendix 17) provided both gauge and absolute values.

Although the operators have been trained on the differences in instruments, this mixed convention provided the potential for misreading or misapplying the data required for the EOIs.

The licensee stated that the use of instruments that give measurements in variant terms had been identified during the detailed control room design review and was the subject of Human Engineering Discrepancy (HED) 0201. The HED was previously scheduled for completion before restart and would result in rescaling of the low-range instruments to units of psig. During the inspection, the licensee informed the team that the HED would likely be reprioritized for completion during the next refueling outage instead of before restart because of problems with obtaining materials for the required modification. The licensee should ensure, through operator training and procedural nomenclature, that this engineering unit difference does not cause confusion.

(8) EOI-2.1.2 specified 160°F drywell temperature as an entry condition for drywell/temperature (DW/T). The initial actions of Step DW/T-1 required only maximizing drywell cooling. Abnormal Operating Instruction (AOI) 2-AOI-64-6, "High Drywell Temperature," Revision 0, Step 4.1.4, required that the reactor be manually scrammed and EOI-2 entered if drywell temperature reached 160°F. The licensee was unable to provide the basis for the AOI requiring a reactor scram and stated that the EOI and AOI would be reviewed and revised as necessary to ensure consistency and a valid basis for the prescribed AOI actions.

(9) The words "Continuous SRV Air Supply" were interpreted by the operators as referring to annunciator 2-PA-3-70 on panel 2-XA-55-3E in carrying out the actions in Step RC/P-2.1. An alarm on this annunciator would be sufficient to prompt action to place safety relief valves in "Close" or "Auto." This step should be modified to allow use of control air and/or containment atmosphere dilution system nitrogen.

(10) The key number specified in Step RC/Q-5.4 for gaining access to panel 9-16 (individual scram switches) in the auxiliary instrument room had been changed from number 192 (EOI) to number 85 (key locker). Finding the proper key in the control room key locker required the operators to conduct a search of the key list for the name of the panel (which was not provided in the procedure) in order to identify the new key storage location. This problem was the same for other keys used in the walkdown of Appendix 9 (key numbers 167 and 168, panel 9-27).

Either the key locker storage locations should be changed to match the as-written EOIs or the EOIs changed to match the key locker storage locations. The EOIs should also include the noun name of the key. A long-term method should be developed to ensure that key locations are not changed until a corresponding change to dependent procedures has been completed. In addition to the key-numbering problem, the team noted that the operators had difficulty with the key sticking in the panel door lock

mechanisms and in turning the key. The licensee should lubricate or repair these locks so that a key will not break under the stressful conditions of an emergency.

(11) A general problem in the Unit 2 control room stemmed from the removal of nearly all operator aids from the control boards. Aids that the operators expressed a strong desire to have reinstalled included the following:

- A general arrangement drawing of the suppression pool, showing the location of safety relief valve (SRV) discharges and temperature detectors, should be posted near SRV controls and torus temperature instruments.

- A drawing showing the general arrangement and location of thermocouples in the drywell should be posted near temperature instrument TI-80-34.

- The suppression pool heat capacity and temperature limit curve should be posted.

- The checklist used for keeping track of the status of system interlock bypasses and entry conditions should be posted in the control room. This checklist was posted in the simulator, but several of the bypasses that could be required during use of the EOIs were missing. The licensee should consider adding the following bypasses to this checklist:

  -- reactor core isolation cooling (RCIC) test mode isolation bypass
  -- RCIC high reactor water level trip bypass
  -- RHR injection valve timers bypass
  -- containment venting bypass

  The licensee stated that a plan existed for replacing the operator aids and that the above specific recommendations would be evaluated.

(12) The directions in the EOI appendices were internally inconsistent in format. Examples of these inconsistencies were the following:

- Valve identification numbers (e.g., FCV-1-14) were used in directions to close or verify closed valves. A different number (e.g., 2-HS-1-14a) was used to identify the corresponding valve control switch handle that must be operated in the control room. The name of the valve was not used in the procedure.

- Only the pump name (e.g., CS Pump A) and not the component or control switch identification number was used in other sections of the procedure directing the verification of run status or the starting of pumps.

- Elsewhere, the appendices provided directions in short-text form (e.g., "Verify HPCI Running").

Directions in an appendix and throughout the EOIs should be made more internally consistent (e.g., both the name and the associated control device number should be used when providing directions for a verification or action step involving the operation of a component).

(13) Appendix 7, Subsection g, required jumpers to be installed for the pressure suppression chamber head tank pumps in reactor motor-operated valve boards 2C and 2B. These jumpers had to be attached to terminals that were very deep within the compartments. Emergency installation of these jumpers would be both difficult and dangerous. Alligator-clip-type jumpers would not attach securely to the terminals, and loosening the lugs for the installation of spade-type jumpers would be dangerous and difficult. The licensee should consider modifications to facilitate easier and safer installation of these jumpers.

Similarly, Agastat timing relays 10A-K45 A and B that were installed in panels 9-32 and 9-33 as specified in Appendix 16 were difficult to reach and were located close to other energized wiring.

Also, the terminal numbers that were referenced in Appendix 8 on wires to the relay terminals inside panels 9-15 and 9-17 (main steam isolation valve interlocks) were extremely small and hard to read. The wires were marked with number tape. Larger or more easily readable wire and terminal numbering should be used.

Some jumpers required to be installed by the appendices on terminal strips had to be connected to terminals with insulated shanks or to terminals that did not protrude far enough from the terminal strip to allow the reliable connection of alligator-clip-type jumpers. The use of spade-type ends on jumpers would provide a solid, reliable connection, but would involve unscrewing the terminal screws to attach the jumpers. Licensee evaluation of a method of jumpering that would be fast and reliable appeared to be warranted.

(14) The team noted that the spoolpiece used to connect auxiliary steam to the RCIC steamline was unsecured on the gratings in the area of valves 71-565 and 12-777 and exposed to flange seating surface or other damage. The licensee should protect such pieces of equipment to ensure they will be available and functional. The team noted that the blank piping flanges were misaligned by about two inches. Discussions with operators confirmed that, although the installation was difficult, the spoolpiece could be installed by two people in less than two hours.

The team noted that the spoolpiece used to connect auxiliary steam to the HPCI steamline was unsecured under debris on the mezzanine near valves 73-587 and 12-778. As in the case of the RCIC spoolpiece, this equipment could also be damaged or discarded. The team noted that the blank flanges where the spoolpiece was to be installed were out of axial alignment by approximately three inches. Discussions with operators and mechanical maintenance personnel confirmed that connecting this spoolpiece would require chainfalls to spring the flanges into alignment and would require four people working about four hours to complete the connection. The licensee should consider realigning the auxiliary steamline with the connection on the HPCI steam piping.

(15) Bay numbers were not shown on the outside of the instrument panels in the auxiliary instrument room. Operators attempting to enter a specific bay to perform an interlock bypass or other local action had to open the panel backs to determine the bay number. Bay numbers were marked on the inside doors of panels with pencil. One relay (16A-K1C, AC) was labeled with a piece of masking tape and a handwritten letter code. The licensee should survey the cabinets for similar labeling problems and take corrective action.

(16) The general state of housekeeping in areas observed during the walkdowns of EOIs (outside the control room) was poor (e.g., the team noted debris around HPCI spoolpieces, debris in auxiliary instrument room panels, etc.).

(17) The BFN staff had placed orange arrows with "EOI" in black letters in auxiliary instrument room panels to direct operators to the locations of relays, buses, and terminals where jumpers and bypasses were called for in the EOI appendices. The "EOI arrows" were beneficial to the operators during the walkdown of the appendices.

3.4 Validation and Verification Program

Six specific problems with the Browns Ferry verification and validation program for EOIs were noted in the NRC technical evaluation report (TER) for the procedure generation package (PGP) dated May 2, 1988. The team reviewed the corrective action taken by the BFN staff pertaining to these problems. Five of the six problems have been addressed through the addition of clarification statements on pages 21 through 24 of the PGP. The problem that had not received corrective action was described as follows in the TER, "Instrument and control details (such as resolution) were absent from the criteria checklist used to evaluate performance during the validation."

The licensee provided the team with the following explanation from PGP Attachment 4:

> Although the word "resolution" was not specifically used in the checklist, certain items in the "Plant Compatibility" section of the checklist were used to determine if control room instrumentation was adequate to support the procedures being validated. We do not feel that any additional information is necessary.

Although the applicable validation procedure did not contain detailed guidance, discussions with operations support personnel showed that validation activities were being conducted to the same standards as those applied during the detailed control room design review (DCRDR) task analysis and survey. The team reviewed the results of these DCRDR-type reviews of portions of EOI-1 and EOI-2 and all of EOI-3 that were performed in April 1988. Since the licensee was meeting the intent of the corrective action applicable to this TER problem, PMI-12.9 should be updated to formalize and require this rigorous approach to ensure that it is applied in future validation efforts.

The correction of the problems noted in the TER through the addition of information in the PGP was satisfactory at BFN. The licensee stated that the PGP will soon become a controlled document.

The team reviewed the two procedures governing verification and validation and identified no additional problems (other than those noted in the TER). These procedures were PMI-12.8, "Verification of Emergency Operating Instructions," Revision 1, and PMI-12.9, "Validation of Emergency Operating Instructions," Revision 2.

The types of problems noted in Sections 3.1 and 3.3 of this report concerning differences between the EOIs and the PSTGs can be partially attributed to the fact that the PSTGs continued to undergo modification after they were used to develop the currently approved revision of the EOIs. Because the PSTGs were not a controlled document, the team could not determine when these changes were made. Because some of the differences between the EOIs and PSTGs cannot be attributed to these post-EOI development revisions, the licensee should increase the level of attention to future verification efforts. The above statement is based on a review by the team of a new draft of the EOIs in which many of the EOI/PSTG differences noted in Sections 3.1 and 3.3 continue to exist.

3.5 Postaccident Reactor Building Habitability and Reentry

The BFN EOIs required entry into the reactor building during and after an accident to perform local operations (e.g., alternate boron injection on standby liquid control system failure and emergency control rod insertion on scram failures).

The team reviewed Emergency Plan Implementing Procedures EPIP-15, "Emergency Exposure," Revision 0, and EPIP-16, "Recovery Procedures," Revision 0, which provided the instructions for personnel reentry into the reactor building, and determined that they included only very basic information regarding maximum dose limits and precautions for reentry. The procedures did not include specific reentry routes for expected EOI operations nor any information regarding anticipated dose rates.

NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used in Post Accident Operations," required that each licensee provide for adequate access to plant areas to permit an operator to aid in the mitigation of or recovery from an accident. This item required the licensee to identify plant areas to which such access was required and to analyze the adequacy of radiation protection based on specific source terms. The licensee's evaluation and status were provided to the NRC in letters dated December 23, 1980, April 14, 1982, and June 17, 1982. Collectively, these letters stated that no modifications to plant shielding were required and that emergency operating procedures did not require reactor building reentry for accident mitigation.

On July 10, 1981 the NRC issued a confirmatory order that required the licensee to implement specific actions regarding Item II.B.2. NRC response and acceptance of the licensee's positions and actions were documented in an NRC safety evaluation report dated March 8, 1983, and NRC Region II Inspection No. 50-260/86-03 conducted in January 1983. The licensee had concluded that, if the NRC-specified source terms were used, the postaccident radiation levels within the reactor building would preclude personnel reentry and stipulated that the plant design

would support all accident operations without requiring reentry. These analyses and conclusions were made during development of the current EOIs and apparently did not consider the reentry requirements in the EOIs.

The team reviewed Drawing Series 47225-100 through -124, "Harsh Environmental Data Drawings," which indicated that in the areas to which access was required the projected radiation levels were extremely high (10E4-10E5 R/hr) using the source terms specified in NUREG-0737.

Discussions with the BFN plant licensing and operations support staff personnel indicated that the reactor building radiation environment was informally considered during the preparation of the EOIs; however, correlation with NUREG-0737, Item II.B.2 data had not been made. The licensee generally considered the NUREG-0737 source terms unrealistically high for all but the most severe accidents and believed that the EOIs had sufficient contingency options to ensure accident mitigation regardless of reactor building reentry restraints. No further licensee action was planned.

3.6 EOI Exercise Using the Plant-Specific Simulator

To ensure that the BFN EOIs could be correctly implemented, the team developed four accident scenarios that were conducted on the BFN simulator using licensed operators. The scenarios were designed to determine whether the EOIs provided the operators with sufficient guidance so that their required actions during an emergency were clearly outlined and could be performed. The scenarios exercised parallel EOI paths and contingency procedures. Hence, the scenarios demonstrated whether: (1) the EOIs caused unnecessary duplication of operator actions, (2) transitions from different EOI paths and contingency procedures could be made satisfactorily, and (3) all the operator actions could be performed concurrently when required.

The scenarios were designed to exercise the maximum number of EOI decision paths and contingency procedures in the available simulator time. The event sequences were accelerated by the use of malfunctions beyond the design bases of the plant but within the scope of the EOIs.

The four scenarios were each conducted by two different operating crews. Each operating crew consisted of a shift operations supervisor, an assistant shift operations supervisor (ASOS), two reactor operators, and a shift technical advisor. The reactor operators at BFN were called unit operators. The ASOS was responsible for reading and directing EOI actions. A discussion of the scenarios follows:

Simulator Scenario No. 1

The first scenario was used by the inspection team to identify the operators' roles and normal methods of communication and to confirm the team member assignments while observing the scenarios. This scenario involved a total loss of feedwater with the reactor at 100-percent power. The subsequent low reactor water level required entry into the EOIs. The scenario was terminated when reactor water level was restored to the normal range using high pressure coolant injection (HPCI).

The inspection team made the following observations:

(1) The ASOS entered and executed EOI-1.

(2) The reactor operator told the ASOS that he was going to control reactor pressure by using the safety relief valves before the ASOS reached that point in the EOIs.

(3) Communications between the operators was generally not very good in that there was no formal command/acknowledgement system.

(4) One crew erroneously believed the plant was in an anticipated transient without scram (ATWS) condition for the first five minutes following a scram in which all the control rods were fully inserted.

Simulator Scenario No. 2

The second scenario was designed to exercise the following EOIs and contingency procedures:

(1) EOI-1, "Reactor Pressure Vessel Control"
(2) C1, "Alternate Level Control"
(3) C2, "Emergency Depressurization"
(4) C3, "Steam Cooling"
(5) C4, "Reactor Flooding"

The transient was initiated from 100-percent reactor power with a loss of feedwater pumps. An ATWS malfunction was inserted that kept all the control rods from moving when low reactor water level generated a scram signal. The HPCI and RCIC systems were prefailed and hence would not inject water into the reactor vessel. Approximately three minutes into the scenario, the control rods were inserted into the core, and a station blackout occurred. This condition left no systems available to inject water into the reactor pressure vessel. At the end of the scenario, one emergency diesel generator was made available and hence an injection system then became available.

The inspection team made the following observations:

(1) The ASOS generally entered and executed the correct EOIs and contingency procedures.

(2) The ASOS incorrectly read a logic diagram in Contingency C1, "Alternate Level Control," and proceeded to Step C7 instead of Step C1-7. The ASOS corrected his error before proceeding with Step C7.

(3) The ASOS did not consistently use the checkoff blanks associated with each step for placekeeping (finding and keeping the correct place in the procedures) as required by management directive.

(4) Neither crew was aware that a station blackout condition had occurred. For example, the ASOS asked about the availability of ac-powered injection systems (core spray and residual heat removal) and the abnormal procedure for station blackout was not entered.

(5) The ASOS made the decision to perform the reactor flooding contingency procedure on the basis of existing plant conditions instead of actually following the EOIs that directed that action.

(6) The reactor operator left three safety relief valves open while he was supposed to be controlling reactor pressure in a band specified in the EOIs. This resulted in an unplanned reactor depressurization before the emergency depressurization step in the EOIs was reached.

Simulator Scenario No. 3

The third scenario was designed to exercise the following EOIs and contingency procedures:

(1) EOI-1, "Reactor Control"
(2) EOI-2, "Containment Control" (including containment venting)
(3) C2, "Emergency Depressurization"
(4) C4, "Reactor Flooding"
(5) C5, "Level/Power Control"

To maximize energy deposition to the primary containment, an ATWS with no control rod movement and a steamline leak in the drywell were initiated from 100-percent reactor power. A main steam isolation valve closure initiated the event. The standby liquid control system was made inoperable. During the subsequent increase in drywell and suppression chamber temperature and pressure, the residual heat removal pumps were failed to preclude use of containment sprays. The simulator computer model was not capable of simulating drywell pressures in excess of 15 psig. To achieve drywell pressures greater than 15 psig, containment parameters were orally provided to the operators and the scenario continued with a static simulator. Continued containment pressurization culminated in containment venting.

The inspection team made the following observations:

(1) The ASOS generally entered and executed the correct EOIs and contingency procedures, which involved simultaneous execution in several areas.

(2) During the first 15 minutes of the scenario, the ASOS could not keep up with all the multiple-action steps in the EOIs that had to be performed concurrently.

(3) One crew did not vent the suppression chamber as required by the EOIs when the suppression chamber air temperature was less that 210°F. Because the ASOS erroneously read drywell temperature instead of suppression pool temperature and found that it was more than 210°F, he assumed venting could not be initiated at that time.

(4) One crew entered the drywell spray initiation curve using drywell temperature instead of suppression chamber air temperature as required by EOI-2. This delayed the attempt to spray the drywell for about 15 minutes.

(5) The presence of a safety parameter display system would have greatly improved the ability of the operators to use the EOIs and manage the emergency. The safety parameter display system is not scheduled for installation until the next operating cycle after restart.

(6) Use of the shift technical advisor (STA) and shift operations supervisor (SOS) during major events needs to be improved. In addition, the roles of the STA and SOS should be better defined to ensure that someone is aware of the overall integrated picture of plant status and the mitigating actions being taken.

(7) Because of time restraints, the ASOS did not consistently use the procedure checkoff blanks.

(8) One crew relied on the reactor operators to evaluate the heat capacity temperature limit and other curves in the EOIs, but the other crew relied on the ASOS to perform these evaluations. The roles and responsibilities of the operators in all crews should be consistent.

Simulator Scenario No. 4

This scenario was designed to exercise the following EOIs and contingency procedures:

(1) EOI-1, "Reactor Pressure Vessel Control"

(2) EOI-3, "Secondary Containment Control and Radioactivity Release Control"

(3) C2, "Emergency Depressurization"

To produce an unisolable leak from the reactor pressure vessel to the secondary containment, a failure of the high pressure coolant injection (HPCI) steam supply line drain pot was initiated. The HPCI steam supply isolation valves failed to close because of malfunctioning breakers. Subsequently, a leak in the reactor water cleanup heat exchanger room produced a second high-temperature area in the secondary containment. Eventually, both area temperatures exceeded their maximum safe operating temperature limits, which required emergency depressurization of the reactor vessel.

The inspection team made the following observation:

(1) The ASOS entered and executed the correct EOIs and contingency procedures. However, during one performance of the scenario, the ASOS did not physically get out the EOIs and read them until approximately seven minutes after the entry condition was met.

3.7 EOI Training

The inspection team reviewed the adequacy of requalification training with respect to the EOIs, use of the plant-specific simulator for EOI training, and methods of training used before EOI revisions were implemented.

3.7.1 Requalification Training

The requalification training program consisted of eight weeks of training per year, including four weeks at the simulator. The training was evenly spaced throughout

the year in 1-week intervals based on a five-shift rotation. Classroom training included lectures on the EOI cautions and bases of the EOI steps. Simulator training consisted of four hours of classroom preparation and four hours on the simulator each day. A wide range of simulator scenarios was used which exercised most of the EOI paths and contingency procedures. The purpose of the scenarios was to ensure proper use and performance of the EOIs. The training scenarios were not as complex as the second and third scenarios used by the inspection team. The shift technical advisor was included in the requalification training. The team concluded that the EOI training was adequate.

3.7.2 Training on EOI Revisions

The requirements for EOI training were contained in Plant Managers Instruction (PMI) 12.6, "Implementation and Maintenance of Emergency Operating Instructions," Revision 2. The PMI stated that the amount and type of operator training required in regard to changes to the EOIs were evaluated on the basis of the magnitude of the change. This evaluation was made by the operations superintendent. Changes to the EOIs were not issued for use in the control room until appropriate operating crew training had been completed. The team concluded the training program for EOI revisions was adequate.

3.8 Licensee Action on Prior NRC Inspection Findings

(Closed) Inspector Followup Item (IFI) 50-260/87-42-01: Confirm that Procedures EOI-3 and EOI-4 are issued and implemented prior to unit restart.

During the previous inspection the team had found that the licensee had developed draft procedures for secondary containment control (EOI-3) and radiation release control (EOI-4), but had not committed to the NRC to have them issued and fully implemented before restart. During the inspection, the team found that the procedures had been issued and personnel training had been provided. Specific team findings regarding these procedures are provided elsewhere in this report.

4.0 EXIT MEETING

On August 18, 1988, the team and other NRC representatives held an exit meeting with licensee personnel and discussed the scope and findings of the inspection. Persons contacted by the team and attendees at the exit meeting are identified in Attachment A. Mr. J. E. Konklin, Chief, Team Integration Section, NRR, represented NRC management at the exit meeting. During the inspection, the team also contacted other members of the licensee's staff to discuss issues and ongoing activities.

ATTACHMENT A
PERSONS CONTACTED
EXIT MEETING ATTENDEES

NAME	ORGANIZATION/TITLE
*J. D. Allen	Site Procedures Group
R. B. Booher	Shift Technical Advisor
*D. Bradley	Division of Nuclear Engineering
*C. Brooks	NRC Resident Inspector
*D. Carpenter	NRC Senior Resident Inspector
*J. Chase	Consultant to TVA
*R. Delay	Division of Nuclear Construction
*R. D. Erickson	Plant Operations Review Staff
*J. W. Hutton	Operations Supervisor
*C. S. Hsieh	Licensing
R. G. Jones	Shift Operations Supervisor (Training)
T. G. Jones	Shift Operations Supervisor
*R. King	Division of Nuclear Construction
G. Little	Systems Engineer (SRO)
T. E. Mayfield	Simulator Training Supervisor
*N. C. McFall	Licensing
*J. Mewbourne	Operations Training
R. Miller	Unit Operator (RO)
R. J. Moll	Lead Simulator Instructor/Operations Training
*H. J. Monroe III	Nuclear Procedures Staff
*B. C. Morris	Licensing
R. W. Moye	Unit Operator (RO)
A. T. Rogers	Shift Technical Advisor
*J. Savage	Licensing
E. R. Scillian	Assistant Shift Operations Supervisor
*T. S. Tracy	Operations Support Consultant
J. Walker	Assistant Unit Operator
J. G. Walker	Plant Manager
E. S. Westfield	Unit Operator (RO)
*W. Williamson	Technical Support Engineer
M. Woloszyn	Unit Operator (RO)

* Denotes those present at the exit meeting on August 18, 1988.

ATTACHMENT B

DOCUMENTS REVIEWED

NUMBER	TITLE	REVISION
AOI-57-2	Station Blackout	1
EOI-1	Reactor Pressure Vessel Control	2
EOI-2	Primary Containment Control	2
EOI-3	Secondary Containment Control and Radioactivity Release Control	2
PMI 12.6	Implementation and Maintenance of Emergency Operating Instructions	2
PMI-12.7	Writer's Guide for Emergency Operating Instructions	2
PMI-12.8	Verification of Emergency Operating Instructions	1
PMI 12.9	Validation of Emergency Operating Instructions	2
PMI 12.12	Conduct of Operations	1
--	BWR Owners Group Emergency Procedure Guidelines, including Appendices A, B, C, and D	3
--	BFN Emergency Procedure Guidelines, including Appendices A, B, C, and D	--
--	BFN Step Deviation Documentation	NA
--	BFN Procedures Generation Package	--

ATTACHMENT C

ABBREVIATIONS

AOI	abnormal operating instruction
ASOS	assistant shift operations supervisor
ATWS	anticipated transient without scram
BWROG	Boiling Water Reactor Owners' Group
BFN	Browns Ferry Nuclear Power Station
DCRDR	detailed control room design review
EOI	emergency operating instruction
EOP	emergency operating procedure
EPGs	emergency procedure guidelines
EPIP	emergency plan implementing procedure
GE	General Electric
HED	human engineering discrepancy
HPCI	high pressure coolant injection
PGP	procedures generation package
PMI	plant managers instruction
PSTG	plant-specific technical guidelines
QA	quality assurance
RCIC	reactor core isolation cooling
RHR	residual heat removal
RHRSW	RHR service water
RX	reactor
SDSP	site director standard practice
SBGT	standby gas treatment
SOS	shift operations supervisor
SRV	safety relief valve
STA	shift technical advisor
TER	technical evaluation report
TI	temporary instruction
TMI	Three Mile Island
TVA	Tennessee Valley Authority
WC	writers' guide

EOI Control Section Designations

RC/L	reactor pressure vessel/level
RC/P	reactor pressure vessel/pressure
RC/Q	reactor pressure vessel/power
DW/T	drywell/temperature
PC/P	primary containment/pressure
SP/T	suppression pool/temperature
SP/L	suppression pool/level
SC/L	secondary containment/level
SC/R	secondary containment/radiation
SC/T	secondary containment/temperature
RR	radioactivity release control