TECHNICAL EVALUATION REPORT

AUXILIARY FEEDWATER SYSTEM AUTOMATIC INITIATION AND FLOW INDICATION (F-16, F-17)

TOLEDO EDISON COMPANY

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

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1. INTRODUCTION

1.1 PURPOSE OF REVIEW

The purpose of this review is to provide a technical evaluation of the emergency feedwater system design to verify that both safety-grade automatic initiation circuitry and flow indication are provided at the Davis-Besse Nuclear Power Plant, Unit 1. In addition, the steam generator level indication available at the Davis-Besse plant is described to assist subsequent NRC staff review.

1.2 GENERIC ISSUE BACKGROUND

A post-accident design review by the Nuclear Regulatory Commission (NRC) after the March 28, 1979 incident at Three Mile Island (TMI) Unit 2 has established that the auxiliary feedwater (AFW) system should be treated as a safety system in a pressurized water reactor (PWR) plant. The designs of safety systems in a nuclear power plant are required to meet general design criteria (GDC) specified in Appendix A of 10CFR50 [1].

The relevant design criteria for the AFW system design are GDC 13, GDC 20, and GDC 34. GDC 13 sets forth the requirement for instrumentation to monitor variables and systems (over their anticipated ranges of operation) that can affect reactor safety. GDC 20 requires that a protection system be designed to initiate automatically in order to assure that acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences. GDC 34 requires that the safety function of the designed system, that is, the residual heat removal by the AFW system, be accomplished even in the case of a single failure.

On September 13, 1979, the NRC issued a letter [2] to each PWR licensee that defined a set of requirements specified in NUREG-0578 [3]. It required that the AFW system have automatic initiation and single failure-proof design consistent with the requirements of GDC 20 and GDC 34. In addition, auxiliary feedwater flow indication in the control room should be provided to satisfy the requirements set forth in GDC 13.

During the week of September 24, 1979, seminars were held in four regions of the country to discuss the short-term requirements. On October 30, 1979, another letter was issued to each PWR licensee providing additional clarification of the NRC staff short-term requirements without altering their intent [4].

Post-TMI analyses of primary system response to feedwater transients and reliability of installed AFW systems also established that, in the long term, the AFW system should be upgraded in accordance with safety-grade requirements. These long-term requirements were clarified in the letter of September 5, 1980 [5]. This letter incorporated in one document, NUREG-0737 [6], all TMI-related items approved by the commission for implementation at this time. Section II.E.1.2 of NUREG-0737 clarifies the requirements for the AFW system automatic initiation and flow indication.

1.3 PLANT-SPECIFIC BACKGROUND

The Toledo Edison Company responded to the NRC requirements in a letter dated September 16, 1981 [7]. This letter provided electrical and piping diagrams as well as other information to supplement the Davis-Besse FSAR.

The review of the AFW system at the Davis-Besse plant began in November 1981, based on the criteria described in Section 2 of this report.

2. REVIEW CRITERIA

To improve the reliability of the AFW system, the NRC required licensees to upgrade the system, where necessary, to ensure timely automatic initiation when required. The system upgrade was to proceed in two phases. In the short term, as a minimum, control-grade signals and circuits were to be used to automatically initiate the AFW system. This control-grade system was to meet the following requirements of NUREG-0578, Section 2.1.7.a [3]:

"1. The design shall provide for the automatic initiation of the auxiliary feedwater system.

2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

3. Testability of the initiating signals and circuits shall be a feature of the design.

4. The initiating signals and circuits shall be powered from the emergency buses.

5. Manual capability to initiate the auxiliary feedwater system from the control room shall be retained and shall be implemented so that a single failure in the manual circuits will not result in the loss of system function.

6. The ac motor-driven pumps and valves in the auxiliary feedwater system shall be included in the automatic actuation (simultaneous and/or sequential) of the loads to the emergency buses.

7. The automatic initiating signals and circuits shall be designed so that their failure will not result in the loss of manual capability to initiate the AFW system from the control room."

In the long term, these signals and circuits were to be upgraded in accordance with safety-grade requirements. Specifically, in addition to the above requirements, the automatic initiation signals and circuits must have independent channels, use environmentally qualified components, have system bypassed/inoperable status features, and conform to control system interaction criteria, as stipulated in IEEE Std 279-1971 [8].

The capability to ascertain the AFW system performance from the control room must also be provided. In the short term, steam generator level indication and flow measurement were to be used to assist the operator in maintaining the required steam generator level during AFW system operation. This system was to meet the following requirements from NUREG-0578, Section 2.1.7.b:

"1. Safety-grade indication of auxiliary feedwater flow to each steam generator shall be provided in the control room.

2. The auxiliary feedwater flow instrument channels shall be powered from the emergency buses consistent with satisfying the emergency power diversity requirements of the auxiliary feedwater system set forth in Auxiliary Systems Branch Technical Position 10-1 of the Standard Review Plan, Section 10.4.9 [9]."

The NRC staff has determined that, in the long term, the overall flowrate indication system for Babcock & Wilcox plants must include at least two safety-grade auxiliary feedwater flowrate indicators for each steam generator. The flowrate indication system should conform to the following salient paragraphs of IEEE Std 279-1971:

- o 4.1 - General Functional Requirements
- o 4.2 - Single Failure
- o 4.3 and 4.4 - Qualification
- o 4.6 - Channel Independence
- o 4.7 - Control and Protection System Interaction
- o 4.9 and 4.10 - Capability for Testing.

The operator relies on steam generator level instrumentation, in addition to auxiliary feedwater flow indication, to determine AFW system performance. The requirements for this steam generator level instrumentation are specified in Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident" [10].

3. TECHNICAL EVALUATION

3.1 GENERAL DESCRIPTION OF AUXILIARY FEEDWATER SYSTEM

The auxiliary feedwater (AFW) system at the Davis-Besse plant supplies water to the secondary side of the steam generators for reactor decay heat removal when normal feedwater sources are unavailable due to loss of offsite power or other malfunctions. The system consists of two steam turbine-driven auxiliary feedwater pumps (1050 gpm) capable of delivering feedwater to either or both steam generators. The AFW system is part of the steam and feedwater rupture control system (SFRCS) and, as stated in the FSAR, is designed in accordance with IEEE Std 279-1971 [8].

3.2 AUTOMATIC INITIATION

3.2.1 Evaluation

Auxiliary feedwater flow to the steam generators is provided by the SFRCS, which is designed to prevent release of high energy steam and to automatically start the AFW system in the event of a main steam line rupture or main feedwater line rupture or when preset levels of any of the following parameters are exceeded:

1. low level in either steam generator
2. main steam line rupture (low-pressure)
3. main feedwater line rupture (high main feedwater/steam generator differential pressure)
4. loss of all four reactor coolant pumps
5. loss of both main feed pumps.

System valves are controlled by the SFRCS. The AFW system consists of two turbine-driven pump trains in parallel. Each train has two motor-operated, normally shut valves in series and one motor-operated, normally shut valve in a cross-connect line which connects between the two series valves on one train and downstream of both series valves on the other train. The cross-connect lines allow either pump to feed either or both steam generators. Open/closed valve position indication is provided in the control room for the motor-operated valve. The AFW system can be controlled automatically by the SFRCS or manually from the control room or auxiliary shutdown panel. The operation of either pump provides the capacity to remove decay heat from the steam generators at a rate sufficient to prevent overpressurization of the reactor coolant system and to maintain steam generator levels. Therefore, the AFW system is capable of automatically initiating appropriate protective action with precision and reliability whenever a condition monitored by the system reaches a preset level.

The primary source of water is the condensate storage tank which is sized so that a total condensate inventory may be available to the pumps sufficient to remove decay heat for 13 hours plus a subsequent cooldown to 280°F. The backup water supplies are the service water system and the fire protection system. Low-pressure switches are provided on the AFW pump suction line which will automatically shift supply to service water system.

The AFW system at the Davis-Besse plant is designed as an engineered safeguards system, the entire system meets Safety Class I criteria, and the automatic initiation signals and circuits comply with the single-failure criterion of IEEE Std 279-1971. A review of initiation logic and wiring diagrams revealed no credible single malfunction that would prevent proper protective action at the system level when required. The diverse signals and redundant channels that provide automatic initiation are physically separated, electrically independent, and powered from emergency buses. In addition, the SFRCS consists of two identical, redundant, and independent channels, and each channel consists of one ac-supplied logic train and one dc-supplied logic train. AFW pump 1-1 steam inlet valve (MS-106) and discharge valves (AF3870 and AF360) are powered by dc (Class 1E, essential power supplies E11C, E11D, and E11E) and AFW pump 1-2 steam inlet valve (MS-107) and discharge valves (AF3872 and AF388) are powered by ac (Class 1E, 480-volt power supplies F11A, F12B, and F12A).

The AFW system can be manually initiated. The turbine-driven pumps can be started either from the control room or the local control panel. The automatic initiating circuits are designed to be electrically independent from valves, and initiate the auxiliary feedwater system when the steam generator pressure exceeds the main feedwater pressure by 197.6 psig."

In Section 7.4.2.3 of the Davis-Besse FSAR, the Licensee states that the design of the SFRCS is in compliance with the requirement of IEEE Std 279-1971. In addition, the Licensee states that no output signals from the SFRCS are used for control functions.

The SFRCS is capable of being tested at power, including system logics, actuation devices, and actuated equipment. A manual testing capability is provided for each input signal to the SFRCS to simulate sensor operation. The system level transmitters, pressure switches, and the differential pressure switches are capable of being independently isolated and, while isolated, simulated process parameters can be applied to check calibration.

Operating and channel bypasses are provided as part of the SFRCS and are designed in accordance with IEEE Std 279-1971, as stated by the Licensee. The FSAR describes these operating and channel bypasses in Section 7.4.1.3.4, as follows:

"7.4.1.3.4 Bypasses

SFRCS includes channel bypasses and operating bypasses.

1. Channel bypass: The only bypasses provided are those on the MCB and ASP for the auxiliary feedwater system. These channel bypasses will permit operator selection of manual control or ICS automatic control. The operation of these switches is under administrative control. The switch position of these bypasses are indicated on the MCB and/or ASP and alarmed in the control room.

2. Operating bypasses: Two out of two logic is provided to allow the operator to bypass each channel to prevent initiation under normal cool down when the main steam line pressure drops below 650 psig. The bypasses are automatically reset by a one out of two logic when the main steam line pressure exceeds 650 psig."

All active components of the AFW system are accessible for inspection during normal station operation. The AFW pumps are tested monthly during station operation, by discharging via the recirculation piping back to the condensate storage tanks.

3.2.2 Conclusion

Based on the evaluation in Section 3.2.1, it is concluded that the initiation signals, logic, and associated circuitry of the AFW system at the Davis-Besse plant comply with the long-term safety-grade requirements of Section 2.1.7.a of NUREG-0578 [3] and the subsequent clarification issued by the NRC, with the exception of the AFW pump power supply diversity requirement, i.e., both AFW pumps are turbine-driven.

3.3 FLOW INDICATION

3.3.1 Evaluation

The performance of the AFW system at the Davis-Besse plant can be assessed by indication of AFW flow, steam generator startup range level, steam generator operating range level, AFW system valve position, and AFW pump status.

The Davis-Besse plant has one AFW flow channel per steam generator which is a differential pressure device located downstream of any cross connect to ensure indication of AFW flow delivered to each steam generator. The power supplies (Class 1E), instrumentation, and displays are designed as safety-grade. The AFW flow indication channels provide no protection function.

In Reference 3, the Licensee provided information to justify having only one safety-grade AFW flow channel per steam generator. The Licensee's statement is as follows:

> "As part of our effort to continuously review and upgrade the Auxiliary Feedwater System (AFWS) for reliability and performance, we have completed an extensive reliability analysis of the Davis-Besse AFWS. Results of this analysis, performed by EDS Nuclear Inc., will be made available to the NRC in the near future. The analysis provides a useful framework for evaluating the impact of AFWS flow indicators on overall plant safety.
>
> Results of the Davis-Besse AFWS reliability analysis indicate that the AFWS flow indicators do not contribute substantially to AFWS reliability. The reasons for this are as follows:
>
> 1. The actuation of the two trains of AFWS is accomplished automatically through safety-grade signals. The automatic action includes

switching to a safety-grade backup water supply (service water supply) should the primary AFWS water supply (condensate storage tanks) fail. Operator action based on control room indication of AFWS flows, is therefore not required.

2. Emergency procedures exist to guide operator actions for mitigating faults should one or both trains of the AFWS fail. The AFWS flow indicators provide only one of several measures of AFWS performance, however. The primary indicator of system performance is steam generator secondary side water level which is monitored and indicated inside the control room by safety grade instrumentation. In addition, separate redundant safety grade level indicators are also available in the cabinet room. Other parameters available to the plant operators for diagnosis of faults and identification of corrective actions include condensate storage tank level, AFW pump speed, AFW pump discharge pressure (all non safety grade) and safety grade steam generator pressure.

3. The steam generator level instrumentation which provides the primary and most important indication of availability of secondary side heat sink is powered from a safety grade power supply redundant from the one supplying the flow instrumentation. Thus, it is highly improbable to lose both the level and existing flow indication caused by a loss of channel power.

4. Other plant design and procedures modifications have been implemented at Davis-Besse subsequent to the TMI-2 event. These modifications have addressed the most significant contributions to AFWS unavailability and have reduced the system unavailability by over an order of magnitude. These modifications include:

   - diverse electric power sources (DC-power supplies) for motor operated valves in one train of the AFWS.

   - redundant and seismically qualified turbine exhausts.

   - administrative procedures to lock in position all manual valves and local control stations and handwheels for motor operated valves in the AFWS.

   - automatic steam generator dual level setpoint control.

   - an emergency procedure to supply water to the steam generator through the main feedwater startup pump should both trains of the AFWS fail.

These modifications have greatly reduced the probability for failure to achieve the AFWS safety function, and have thereby diminished the relative significance of the AFWS flow indication.

5. The inclusion of one AFWS flow indicator per train has little impact on the reliability analysis results. The inclusion of a second flow indicator per train has even less impact. The only application for the flow indicator in the reliability analysis is to assist the operator in recovering from a faulted system condition. Generally, the ability to recover is dominated by human factors, not by instrumentation performance. A second flow indicator would not address the more dominant human factors, and would therefore not significantly improve system reliability.

6. The failure of a flow indicator, in itself, will not lead to adverse consequences. It should be noted that the steam generator level and pressure indications provide adequate information for operator to ensure availability of adequate secondary heat sink. Even in the event of a steam line break when the two AFW pumps are feeding the unaffected steam generator, the availability of (unaffected) steam genearator has been previously shown to be acceptable for decay heat removal purposes. There are no credible occurrences in which automatic or manual actions would be taken to terminate AFW flow based on a faulty indication of that flow.

The insensitivity of the overall AFWS reliability to a second AFWS flow indicator and the inclusion of truly diverse indication of the primary performance parameter (steam generator level) make unnecessary a second flow indicator in each train."

Even though, as the Licensee has indicated, other indication and automatic features exist to assist the operator, the availability of two AFW flow channels for each Babcock & Wilcox steam generator is essential. The single AFW flow channel proposed by Toledo Edison does not meet the requirements of NUREG-0737, in particular the single failure requirements.

3.3.2 Conclusion

The present AFW flow indication does not meet the long-term, safety-grade requirements of NUREG-0578, Section 2.1.7.b, and the subsequent clarification issued by the NRC. In order to comply with these requirements, a second safety-grade AFW flow indication channel per steam generator should be installed.

3.4 DESCRIPTION OF STEAM GENERATOR LEVEL INDICATION

There are three level channels for monitoring steam generator level at the Davis-Besse plant. These include: (1) startup range, (2) operate range, and (3) full range level instrumentation. Following is a grouping of these level instrumentations and an indication of their power supply. The instrumentation used for control or automatic initiation of auxiliary feedwater is indicated by an asterisk (*). Under the power source column, some of the instruments are shown to be powered from uninterruptible buses YAU (and YBU) or vice versa. At present, these instruments are powered from power supplies shown outside the parenthesis. Following the modifications scheduled for implementation in the 1982 refueling outage, these instruments will be able to be powered from both sources.

Instrument Number	Service Description	Power Source
Startup Range Instrumentation		
HS-SP9B2	SG 1 SU Level Selector	YAU and YBU
LT-SP9B3	SG 1 SU Level Transmitter*	Y1
LY-SP9B3	SG 1 SU Level for AFPT 1*	Y1
LC-SP9B3	SG 1 SU Level for AFPT 1*	Y1
LI-SP9B3	SG 1 SU Level for Indicator	Y1
LT-SP9B4	SG 1 SU Level Transmitter*	Y2
LY-SP9B4	SG 1 SU Level for AFPT 2*	Y2
LC-SP9B4	SG 1 SU Level for AFPT 2*	Y2
LY-SP9B6	SG 1 SU Level for Buffer Module	Y2
LS-SP9B	SG 1 SU Level for Low Alarm	YBU and YAU
LI-SP9B1	SG 1 SU Level Indicator	Y1
HS-SP9A2	SG 2 Level Selector	YAU and YBU
LT-SP9A3	SG 2 SU Level Transmitter*	Y2
LY-SP9A3	SG 2 SU Level for AFPT 2*	Y2
LC-SP9A3	SG 2 SU Level for AFPT 2*	Y2
LI-SP9A3	SG 2 SU Level Indicator	Y2
LT-SP9A4	SG 2 SU Level Transmitter*	Y1
LY-SP9A4	SG 2 SU Level for AFPT 1*	Y1
LC-SP9A4	SG 2 SU Level for AFPT 1*	Y1
LY-SP9A6	SG 2 SU Level Buffer Module	Y1
LS-SP9A	SG 2 SU Level Low Alarm	YBU and YAU
LI-SP9A1	SG 2 SU Level Indicator	Y2

Instrument Number	Service Description	Power Source
LT-SP9B6	SG 1 Level Transmitter for SFRCS (CH 2)*	Y2
LI-SP9B6	SG 1 Level Indicator for SFRCS (CH 2)	Y2
LSLL-SP9B6	SG 1 Low Level Trip for SFRCS (CH 2)*	Y2
LT-SP9B7	SG 1 Level Transmitter for SFRCS (CH 2)*	D2P
LI-SP9B7	SG 1 Level Indicator for SFRCS (CH 2)	D2P
LSLL-SP9B7	SG 1 Low Level Trip for SFRCS (CH 2)*	D2P
LT-SP9B8	SG 1 Level Transmitter for SFRCS (CH 1)*	Y1
LI-SP9B8	SG 1 Level Indicator for SFRCS (CH 1)	Y1
LSLL-SP9B8	SG 1 Low Level Trip for SFRCS (CH 1)*	Y1
LT-SP9B9	SG 1 Level Transmitter for SFRCS (CH 1)*	D1P
LI-SP9B9	SG 1 Level Indicator for SFRCS (CH 1)	D1P
LSLL-SP9B9	SG 1 Low Level Trip for SFRCS (CH 1)*	D1P
LT-SP9A6	SG 2 Level Transmitter for SFRCS (CH 1)*	Y1
LI-SP9A6	SG 2 Level Indicator for SFRCS (CH 1)	Y1
LSLL-SP9A6	SG 2 Low Level Trip for SFRCS (CH 1)*	Y1
LT-SP9A7	SG 2 Level Transmitter for SFRCS (CH 1)*	D1P
LI-SP9A7	SG 2 Level Indicator for SFRCS (CH 1)	D1P
LSLL-SP9A7	SG 2 Low Level Trip for SFRCS (CH 1)*	D1P
LT-SP9A8	SG 2 Level Transmitter for SFRCS (CH 2)*	Y2
LI-SP9A8	SG 2 Level Indicator for SFRCS (CH 2)	Y2
LSLL-SP9A8	SG 2 Low Level Trip for SFRCS (CH 2)*	Y2
LT-SP9A9	SG 2 Level Transmitter for SFRCS (CH 2)*	D2P
LI-SP9A9	SG 2 Level Indicator for SFRCS (CH 2)	D2P
LSLL-SP9A9	SG 2 Low Level Trip for SFRCS (CH 2)*	D2P
Operate Range Level Instrumentation		
HS-SP9B1	SG 1 Operate Level Selector	YAU (and YBU)
LT-SP9B1	SG 1 Operate Level Transmitter	YBU (and YAU)
LY-SP9B1	SG 1 Operate Level Temperature Compensator	YBU (and YAU)
LS-SP9B1	SG 1 Operate Level High Level Alarm	YBU and YAU
LT-SP9B2	SG 1 Operate Level Transmitter	YAU (and YBU)
LY-SP9B2	SG 1 Operate Level Temperature Compensator	YAU (and YBU)
LRS-SP9B	SG 1 Operate Level Recorder	YBU (and YAU)
HS-SP9A1	SG 2 Operate Level Selector	YAU (and YBU)

Instrument Number	Service Description	Power Source
LT-SP9A1	SG 2 Operate Level Transmitter	YBU (and YAU)
LY-SP9A1	SG 2 Operate Level Temperature Compensator	YBU (and YAU)
LS-SP9A1	SG 2 Operate Level High Level Alarm	YBU and YAU
LT-SP9A2	SG 2 Operate Level Transmitter	YAU (and YBU)
LY-SP9A2	SG 2 Operate Level Temperature Compensator	YAU (and YBU)
LRS-SP9A	SG 2 Operate Level Recorder	YBU (and YAU)

Full Range Level Instrumentation

LI-SP9B2	SG 1 Full Range Level Indicator	YBU (and YAU)
LT-SP9B5	SG 1 Full Range Level Transmitter	YBU (and YAU
LY-SP9B5	SG 1 Full Range Level E/I Converter	YAU
LS-SP9B5	SG 1 Full Range Level Alarm	YAU
LI-SP9A2	SG 2 Full Range Level Indicator	YBU (and YAU)
LT-SP9A5	SG 2 Full Range Level Transmitter	YBU (and YAU)
LY-SP9A5	SG 2 Full Range Level E/I Converter	YBU
LS-SP9A5	SG 2 Full Range Level Alarm	YBU

4. CONCLUSIONS

Based on the evaluation in Section 3.2.1, it is concluded that the initiation signals, logic, and associated circuitry of the AFW system at the Davis-Besse plant comply with the long-term safety-grade requirements of Section 2.1.7.a of NUREG-0578 [3] and the subsequent clarification issued by the NRC, with the exception of the AFW pump power supply diversity requirement, i.e., both AFW pumps are turbine-driven.

The present AFW flow indication does not meet the long-term, safety-grade requirements of NUREG-0578, Section 2.1.7.b, and the subsequent clarification issued by the NRC, because only one safety-grade channel of AFW flow indication is installed. In order to meet the NRC requirements, a second safety-grade AFW flow indication channel per steam generator should be installed.

5. REFERENCES

1. Code of Federal Regulations, Title 10, Office of the Federal Register, National Archives and Records Service, General Services Administration
Revised January 1, 1980

2. NRC, Generic letter to all PWR licensees regarding requirements resulting from Three Mile Island Accident
September 13, 1979

3. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations"
USNRC, July 1979

4. NRC, Generic letter to all PWR licensees clarifying lessons learned short-term requirements
October 30, 1979

5. NRC, Generic letter to all PWR licensees regarding short-term requirement resulting from Three Mile Island accident
September 5, 1980

6. NUREG-0737, "Clarification of TMI Action Plan Requirements"
USNRC, November 1980

7. R. P. Crouse (Toledo Edison)
Letter to J. F. Stolz (NRC)
Subject: Additional Information on AFW System Automatic Initiation
Toledo Edison Company, 16-Sep-81

8. IEEE Std 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, Inc., New York, New York

9. NUREG-75/087, Standard Review Plan, Section 10.4.9, Rev. 1, USNRC, no date

10. Regulatory Guide 1.97 (Task RS 917-4), "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following and Accident," Rev. 2, USNRC, December 1980