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FIGURES

1A.8 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS (NUREG-0737 Item I.C.1) (Cont'd)

Response (Cont'd)

2. NEDO-24708A, Revision 1, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," December, 1980. This report was issued via the letter from D. B. Waters (BWR Owners' Group) to D. G. Eisenhut (NRC) dated March 20, 1981.

3. BWR Emergency Procedure Guidelines (Revision 0) - submitted in prepublication form June 30, 1980.

4. BWR Emergency Procedure Guidelines (Revision 1) - Issued via the letter from D. B. Waters (BWR Owners' Group) to D. G. Eisenhut (NRC) dated January 31, 1981.

5. BWR Emergency Procedure Guidelines (Revision 2) - submitted in prepublication form June 1, 1982, Letter BWROG 8219 from T. J. Dente (BWR Owners Group) to D. G. Eisenhut (NRC).

b. Adequacy of Submittals

The submittals described in Paragraph (a.) have been discussed and reviewed extensively among the BWR Owners' Group, the General Electric Company, and the NRC Staff. The NRC Staff has found (NUREG-0737, p. I.C.1-3) that "the analysis and

1A.8 GUIDANCE FOR THE EVALUATION AND DEVELOPMENT OF PROCEDURES FOR TRANSIENTS AND ACCIDENTS (NUREG-0737 Item I.C.1) (Cont'd)

Response (Cont'd)

guidelines submitted by the General Electric Company (GE) Owners' Group...comply with the requirements (of the NUREG-0737 clarifications)." In Reference 9, the Director of the Division of Licensing states, "we find the Emergency Procedure Guidelines acceptable for trial implementation (on six plants with applications for operating licenses pending)."

GE believes that in view of these findings, no further detailed justification of the analyses or guidelines is necessary at this time.

Emergency procedures developed from the emergency procedures guidelines will be prepared by each applicant and implemented prior to fuel loading. Section 15D.2.3 also addresses the emergency procedure guidelines with regard to their relation to severe accidents. The emergency procedures training program will be made available for review by the NRC by the applicant.

1A.9 SHIFT RELIEF AND TURNOVER PROCEDURES (NUREG-0737 Item I.C.2) (Cont'd)

NRC Position*

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

a. A checklist shall be provided for the oncoming and offgoing control room operators and the oncoming shift supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist.

   1. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   2. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console. (What to check and criteria for acceptable status shall be included on the checklist)

   3. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the

1A.9 SHIFT RELIEF AND TURNOVER PROCEDURES (NUREG-0737 Item I.C.2) (Cont'd)

NRC Position* (Cont'd)

Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

b. Checklist or logs shall be provided for completion by the offgoing and ongoing auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by themselves could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist); and

c. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure (for example, periodic independent verification of system alignments).

Response

The response to this requirement will be supplied by the applicant.

*This position statement is repeated from Reference 10 since it was not provided in detai in either NUREG-0660 or NUREG-0737.

1A.19 REACTOR COOLANT SYSTEM VENTS (NUREG-0737 Item II.B.1)

NRC Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

Each license shall provide the following information concerning the design and operation of the high point vent system:

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for loss-of-coolant accidents initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10 CFR 50.46.

1A.19 REACTOR COOLANT SYSTEM VENTS (NUREG-0737 Item II.B.1) (Cont'd)

NRC Position (Cont'd)

(2) Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Response

The capability to vent the 238 Nuclear Island reactor coolant system is provided by the safety relief valves and reactor coolant vent line as well as other systems. The capability of these systems and their satisfaction of Item II.B.1 is discussed below.

The 238 Nuclear Island design is provided with nineteen power-operated safety-grade relief valves which can be manually operated from the control room to vent the reactor pressure vessel. The point of connection to the main steamlines which exit near the top of the vessel to these valves is such that accumulation of gases above that point in the vessel will not affect removal of gases from the reactor core region.

These power-operated relief valves satisfy the intent of the NRC position. Information regarding the design, qualification, power source, etc., of these valves is provided in Subsection 5.2.2.

1A.19 REACTOR COOLANT SYSTEM VENTS (NUREG-0737 Item II.B.1) (Cont'd)

Response (Cont'd)

Under most circumstances, no selection of vent path is necessary because the relief valves (as part of the automatic depressurization system), HPCS, and RCIC will function automatically in their designed modes to ensure adequate core cooling and provide continuous venting to the suppression pool.

Analyses of water inventory-threatening events with very severe degradations of system performance have been conducted. These were submitted by GE for the BWR Owners' Group to the NRC Bulletins and Orders Task Force on November 30, 1979 (Reference 24). The fundamental conclusion of those studies was that if only one ECC system is injecting into the reactor, adequate core cooling would be provided and the production of large quantities of hydrogen was avoided. Therefore, it is not desirable to interfere with ECCS functions to prevent venting.

The emergency procedure guidelines emphasize the use of HPCS/RCIC as a first line of defense for inventory-threatening events which do not quickly depressurize the reactor. If these systems succeed in maintaining inventory, it is desirable to leave them in operation until the decision to proceed to cold shutdown is made. Thus, the reactor will be vented via RCIC turbine steam being discharged to the suppression pool. Termination of this mode of venting could also terminate inventory makeup if the HPCS had failed also. This would necessitate reactor depressurization via the SRV, which of course is another means of venting.

1A.19 REACTOR COOLANT SYSTEM VENTS (NUREG-0737 Item II.B.1) (Cont'd)

Response (Cont'd)

If the HPCS/RCIC are unable to maintain inventory, the emergency procedures guidelines call for use of ADS or manual SRV actuation to depressurize the reactor so that the low-pressure LPCI and/or LPCS systems can inject water. Thus, the reactor would be vented via the SRV to the suppression pool. Termination of this mode of venting is not recommended. It is preferrable to remain unpressurized; however, if inventory makeup requires HPCS or RCIC restart, that can be accomplished manually by the operator. It is more desirable to establish and maintain core cooling than to avoid venting. If the HPCS/RCIC and safety/relief valves are not operable (a very degraded and extremely unlikely case), another emergency means of venting the reactor must be used. It is emphasized, however, that such emergency venting would be in the interest of core cooling and therefore would be employed under emergency procedure guidelines.

It is thus concluded that there is no reason to interfere with ECCS operation to avoid venting. It is further concluded that the emergency procedure guidelines, by correctly specifying operator actions for RCIC and SRV operation, also correctly specify operator actions to vent the reactor.

In the event of HPCS failure and continued vessel pressurization, the effect of noncondensibles in the RCIC turbine steam was evaluated for three cases:

1A.19 REACTOR COOLANT SYSTEM VENTS (NUREG-0737 Item II.B.1) (Cont'd)

Response (Cont'd)

a. Continuous evolution of noncondensibles due to radiolysis;

b. Quasi-continuous evolution of noncondensibles due to core heatup;

c. The presence of a quantity of noncondensibles in the reactor at the time of RCIC startup.

Case a is a normal operating mode for RCIC and is of no concern.

For Case b to exist, the core must be uncovered. Such a condition requires multiple failures as shown in the degraded cooling analyses. Core uncovery is prevented (or cladding heatup into the rapid oxidation range is prevented) when only one ECC system is operating. For a small pipe break or a loss of feedwater, which would allow the reactor to remain at pressure, the HPCS and/or RCIC pumps would maintain inventory and there would be no substantial hydrogen production. If neither HPCS nor RCIC could maintain inventory, the reactor would be automatically or manually depressurized via safety/relief valves (or via the break, for larger breaks). Low pressure water injection systems (LPCI or LPCS) would then make up inventory. With the core covered neither the rapid generation of noncondensibles nor their accumulation would be possible.

1A.19 REACTOR COOLANT SYSTEM VENTS (NUREG-0737 Item II.B.1) (Cont'd)

Response (Cont'd)

The performance of RCIC under Case c is of concern only if there has been a very substantial production of hydrogen due to core uncovery and there is a need to start the RCIC. This is extremely unlikely and an intolerable circumstance, because it could arise only if the core were allowed to remain uncovered for a long period with the reactor at high pressure. Automatic depressurization system operation and explicit operating instructions and the emergency procedures guidelines are intended to preclude this. If the level has fallen with the reactor at high pressure, the vessel would be depressurized via the relief valves automatically or manually to permit low-pressure injection independent of RCIC performance.

In the post-LOCA condition, it is possible to have noncondensible gases come out of solution while operating the residual heat removal (RHR) system in the shutdown cooling or steam condensing mode of operation. These gases would accumulate at the top of the RHR heat exchanger since this is a system high point and an area of relatively low flow. Gases trapped here will be vented through a 3/4-inch vent line with two safety-related Class 1E motor-operated valves operated from the control room (as shown in Figure 5.4-12). As this vent line and associated valves are part of the original design, they have also been considered in the design-basis accident analysis contained elsewhere in this document. To accommodate the continuous release of noncondensibles

1A.19 REACTOR COOLANT SYSTEM VENTS (NUREG-0737 Item II.B.1) (Cont'd)

Response (Cont'd)

from the RHR Heat Exchanger when employed in the steam-condensing mode, these remote vent valves on the heat exchanger vent line are opened to discharge through a submerged line into the drywell portion of the suppression pool.

Because the relief valves and RCIC will vent the reactor continuously, and because containment hydrogen calculations in normal safety analysis calculations assume continuous venting, no special analyses are required to demonstrate "that the direct venting of noncondensible gases with perhaps high hydrogen concentrations does not result in violation of combustible gas concentration limits in containment."

Conclusion and Comparison with Requirements

The conclusions from this vent evaluation are as follows:

a. Reactor vessel head vent valves exist to relieve head pressure (at shutdown) to the drywell via remote operator action.

b. The reactor vessel head is continuously swept to the main condenser and can be vented during operating conditions via the SRV's to the suppression pool.

c. The RCIC system provides an additional vent pathway to the suppression pool.

1A.19 REACTOR COOLANT SYSTEM VENTS (NUREG-0737 Item II.B.1) (Cont'd)

Response (Cont'd)

d. The size of the vents is not a critical issue because BWR SRV's have substantial capacity, exceeding the full power steaming rate of the nuclear boiler.

e. The SRV's vent to the containment suppression pool, where discharged steam is condensed without causing a rapid containment pressure/temperature transient.

f. The SRV's are not smaller than the NRC defined small LOCA. Inadvertent actuation is a design-basis event and a demonstrated controllable transient.

g. Inadvertent actuation is of course undesirable, but since the SRV's serve an important protective function, no steps such as removal of power during normal operation, should be taken to prevent inadvertent actuation.

h. A dual indication of SRV position (pressure and temperature) is provided in the control room.

i. Each SRV is remotely operable from the control room.

j. Each SRV is seismically and Class 1E qualified

k. Block valves are not required, so block valve qualifications are not applicable.

1A.23 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2) (NUREG-0737 Item II.D.1)

NRC Position

Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

Response

A generic test program has been conducted through the BWR Owners Group to satisfy this requirement. The testing requirement to qualify SRVs for the "expected operating conditions" associated with design-basis accidents and operational transients was determined through systematic analysis of those events as defined in Regulatory Guide 1.70, Revision 2. The conclusion from that evaluation was submitted to the NRC in September 1980 (Reference 25) in response to Item 2.1.2 of NUREG-0578 (Reference 12); the conclusion was that "there is no design-basis accident or transient which requires safety, relief, or dual function SRV's to pass two-phase or liquid flow at high pressure." This submittal, however, acknowledged the alternate shutdown cooling mode which is considered in the design analysis of plants and committed to testing SRV's with liquid under low pressure conditions associated with this event. Additional justification for the conclusion that no high pressure liquid or two-phase discharge testing is

1A.23 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2) (NUREG-0737 Item II.D.1) (Cont'd)

Response (Cont'd)

required was provided by the BWR Owners' Group to the NRC Staff during meetings on February 10, 1981 and March 10, 1981.

A test plan which describes the test program for SRV testing for the alternate shutdown mode of cooling was included in the September 1980 (Reference 25) submittal to the NRC. The purpose of the test plan was two-fold:

a. To demonstrate the capability of each type of SRV used in BWRs to operate satisfactorily under the bounding case of expected water discharge release of low-pressure water with resultant typical SRV discharge pipe loads on the SRV.

b. To measure the SRV piping discharge loads during water discharge through these valves.

The Dikkers 8 x 10 direct-acting SRV to be used in the 238 Nuclear Island was included in this test program.

The test program provides for consideration of remote manual initiation of the SRV's. Among other tests, the program involved the admission of slightly subcooled water at approximately 250 psig for fluid flow testing. This test followed a normal steam discharge test. This sequence of steam test - water test was repeated three times for each valve.

1A.23 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2) (NUREG-0737 Item II.D.1) (Cont'd)

Response (Cont'd)

The acceptance criteria included proper opening on demand (inlet pressure at setpoint pressure); proper blowdown, i.e., SRV does not reclose except when inlet pressure drops below the setpoint minus the blowdown decrement; SRV to open properly on command for relief function; and pressure integrity of the valve body, connections, and piping is maintained at all times.

The generic test program has been completed and final test results were transmitted in a letter from T. J. Dente (BWR Owners' Group) to D. G. Eisenhut (NRC), dated July 1, 1981. The results showed that all of the test criteria were met for all valves tested.

As part of the response to this requirement, it was determined that the high drywell pressure inhibit of the HPCS high level trip on the 238 Nuclear Island should be removed, as it's removal decreases the probability of water entering the steam lines and the SRVs being subjected to high pressure water flow. The following paragraphs describe this modification as it relates to the 238 Nuclear Island design.

Current Design

When a low reactor water level (L2) occurs in the previously defined 238 Nuclear Island design (Figure 7.3-1b), the L2 trip units will provide the one

1A.23 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2) (NUREG-0737 Item II.D.1) (Cont'd)

Response (Cont'd)

out of two, twice logic that initiates HPCS injection into the reactor vessel (5301 initiation signal). When the high drywell pressure inhibit signal is not present, this injection will continue until a high reactor water level (L8) is reached, and then the L8 trip units will provide the two out of two, once logic which will terminate the HPCS injection automatically (5302 termination signal).

If the high drywell pressure inhibit signal is present, the L8 trip is inhibited from initiating, and termination of the HPCS injection will not occur automatically. This high drywell pressure inhibit signal of the HPCS high water level trip can occur under LOCA conditions and non-LOCA conditions, such as a consequence of loss of drywell coolers, or a recirculation pump seal failure. Since the HPCS has the capability of raising the reactor vessel water level to an elevation that is above the main steamline nozzle, this inhibiting action can result in an increased probability of high pressure liquid entering the main steamlines, safety/ relief valves (SRV) and being discharged via the SRV discharge lines to the suppression pool.

Design Modification

Figure 1A.23-1 shows the primary functional change which is being implemented to remove the high drywell pressure inhibit logic. The removal of this inhibit

1A.23 PERFORMANCE TESTING OF BOILING-WATER REACTOR AND PRESSURIZED-WATER REACTOR RELIEF AND SAFETY VALVES (NUREG-0578, SECTION 2.1.2) (NUREG-0737 Item II.D.1) (Cont'd)

Response (Cont'd)

logic assures that the L8 trip will function automatically to terminate the HPCS injection into the reactor vessel when the L8 trip is present.

Separation of the mechanical divisions for initiation of the high water level trip is required to ensure that a single failure of an instrument line does not cause an inadvertent trip at the HPCS system. In order to accommodate this change, the Level 8 trip unit for B21-N674C is reassigned to another division as shown in Figure 1A.23-2.

The plant modifications for the HPCS trip logic described above will be reflected in Figure 7A.3-1 and Figure 5.1-3c following staff approval of this response.

1A.24 DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION (NUREG-0737, Item II.D.3)

NRC Position

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

Response

The 238 Nuclear Island will be equipped with a Safety Relief Valve Open/Closed Monitor (SRVOCM) in order to provide the operator with positive indication of valve position (closed or not closed). A positive system providing status information on the safety/relief valves is required by NUREG-0737 Item II.D.3 to assist the plant operators during normal and abnormal operating conditions by providing the following information:

1. Positive indication of valve position including the "stuck-open" valve condition.

2. Positive identification of the specific valve or valves which are open.

3. Annunciation of activation of the Automatic Depressurization System (ADS) in the control room.

Providing prompt indication and annunciation of valve opening and identification of the valve, enables plant operators to initiate appropriate action in a timely manner.

1A.24 DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION (NUREG-0737, Item II.D.3) (Cont'd)

Response (Cont'd)

As shown in Figure 1A.24-1, the Safety/Relief Valve Positive Open/Closed Position Monitor System consists of three pressure switches connected by a hydraulic sensing line to the discharge piping of the Safety/Relief Valve. The output of each pressure switch is connected to a circuit board assembly containing relays and electronic logic. The circuit board(s) are mounted in a metal NEMA-4 cabinet enclosure at a point outside the containment, remote from the pressure switch.

An open S/RV pressurizes the discharge line and the hydraulic sensing line to the pressure switch, actuating the pressure switch. The electrical output of the pressure switch controls a sensor relay mounted on the circuit board at a point remote from the pressure switch. The relay contacts provide input to the annunciator in the control room, to the process computer and to an indicator light on a control room instrument panel. The pressure switches are designed for LOCA conditions. The pressure switches are activated by increasing pressure in the discharge line and will reset after the S/RV closes and pressure in the line decays. The pressure switches are mounted inside the containment and the circuit board and cabinet are mounted outside the containment in a convenient location. Each relay of the circuit board has isolated testability.

1A.24 DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION (NUREG-0737, Item II.D.3) (Cont'd)

Response (Cont'd)

The system is rated for maximum vessel pressure. A leaking S/RV will not actuate the pressure switch, thus, the system detects only an appreciably open SRV to eliminate misleading indications to the plant operators.

The pressure switches are commercial grade components designed for application inside the containment structure and capable of operation under LOCA environmental conditions. The pressure switches are designed for conditions in excess of the LOCA conditions of 340°F maximum temperature (6-hour period), 200°F continuous operation at 100% relative humidity and 2.5G acceleration on all axes. These pressure switches are fabricated from materials which are compatible with the LOCA environmental conditions. The only nonmetallic materials are in the body of the microswitch which is designed for application in conditions which exceed the LOCA environment.

The circuit board assembly consists of a prewired board with sensor relays mounted. The circuit boards are mounted in a NEMA-4 type cabinet enclosure which has a maximum capacity of eight circuit board modules. The three-channel system provides a 2-out-of-3 logic. With this logic, failure of a single pressure switch will not compromise the reliability of the system and will not result in an erroneous valve position signal to the control room.

1A.24 DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION (NUREG-0737, Item II.D.3) (Cont'd)

Response (Cont'd)

In the three-channel system, shown in Figure 1A.24-1, a single hydraulic sensing line from a SRV discharge pipe is connected by a manifold to three identical pressure switches in parallel. The pressure switches are electrically connected to the sensor relays in the circuit board assembly. The contacts of the relays are wired in a 2-out-of-3 logic which signals the annunciator, computer and indicator lights.

These modifications to the 238 Nuclear Island will satisfy the requirements of NUREG-0737 Item II.D.3 and will be reflected in Figure 5.1-3a and Section 5.2 following staff approval of this response.

A diverse measurement for indication of SRV opening or long-term leakage is provided via temperature elements mounted in thermowells on each of the SRV blowdown pipes to the suppresion pool. These indications provide confirmation of the SRVOCM readouts.

1A.39 REACTOR VESSEL LEVEL INSTRUMENTATION (NUREG-0737 Item II.K.1.23)

NRC Position*

For boiling water reactors, describe all uses and types of reactor vessel level indication for both automatic and manual initiation of safety systems. Describe other instrumentation that might give the operator the same information on plant status. See Bulletin 79-08, Item 4.

Response

The water level measurement for the 238 Nuclear Island is fully described in NEDO-24708A, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors." An outline of this description is provided in the following paragraphs.

Figure 7.7-1 illustrates the reactor vessel elevations covered by each water-level range. The instruments that sense the water level are differential pressure devices calibrated to be accurate at a specific vessel pressure and liquid temperature condition. The following is a description of each water-level range.

a. Shutdown water-level range: This range is used to monitor the reactor water-level during the shutdown condition when the reactor system is flooded for maintenance and head removal. The water-level measurement design is the condensate reference chamber leg type that is not compensated for changes in density. The vessel temperature and

1A.39 REACTOR VESSEL LEVEL INSTRUMENTATION (NUREG-0737 Item II.K.1.23) (Cont'd)

Response (Cont'd)

pressure conditions that are used for the calibration are 0 psig and 120°F water in the vessel. The two vessel instrument penetrations elevations used for this water-level measurement are located at the top of the RPV head and the instrument tap just below the bottom of the dryer skirt.

b. Upset water-level range: This range is used to monitor the reactor water when the level of the water goes off the narrow-range scale on the high side. The design and vessel tap location are the same as outlined above. The vessel pressure and temperature conditions for accurate indication are at the normal operating points.

c. Narrow water-level range: This range uses for its RPV taps the elevation near the top of the dryer skirt and the tap at an elevation near the bottom of the dryer skirt. The instruments are calibrated to be accurate at the normal operating points. The water-level measurement design is the condensate reference chamber type, is not density compensated, and uses differential pressure devices as its primary elements. The feedwater control system uses this range for its water-level control and indication inputs.

1A.58 SEPARATION OF HIGH-PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEM INITIATION LEVELS -- ANALYSIS AND IMPLEMENTATION (NUREG-0737 Item II.K.3.13) (Cont'd)

Response (Cont'd)

plant experience was evaluated to estimate the frequency of occurrence of HPCS* and RCIC initiations. Based on this evaluation, it was concluded that the current design is satisfactory, and a significant reduction in thermal cycles is not achievable or necessary.

Evaluation of Proposed Auto-Restart of RCIC

The BWR Owners' Group sponsored a program to evaluate this concern and develop an appropriate modification. The results of this program were submitted to the NRC via a letter from D. B. Waters, Chairman of BWR Owners' Group, to D. G. Eisenhut, Director of NRC, dated December 29, 1980 (Reference 20). These results conclude that automatic restart of the RCIC would contribute to improved system reliability and that it could be accomplished without adverse effects on system function and plant safety. Therefore, the 238 Nuclear Island design will be modified to allow automatic restart of the RCIC system following its trip on high RPV water level.

*The HPCS system replaces the HPCI system in the 238 Nuclear Island. The above referenced BWR Owners' Group analysis addresses the use of both systems.

1A.58 SEPARATION OF HIGH-PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEM INITIATION LEVELS -- ANALYSIS AND IMPLEMENTATION (NUREG-0737 Item II.K.3.13) (Cont'd)

Response (Cont'd)

The plant modifications to allow automatic restart of the RCIC system following its trip on high RPV water level will be reflected in Subsection 5.4.6 following staff approval of this response. A technical description of this modification is included in Section 15D.2.1.3.1.

1A.61 REPORT ON OUTAGES OF EMERGENCY CORE-COOLING SYSTEMS LICENSEE REPORT AND PROPOSED TECHNICAL SPECIFICATION CHANGES (NUREG-0737 Item II.K.3.17)

NRC Position

Several components of the emergency core-cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Response

The response to this requirement will be supplied by licensees of plants with sufficient operating time to provide useful data.

This requirement does not apply to NTOL plants or any future plants.

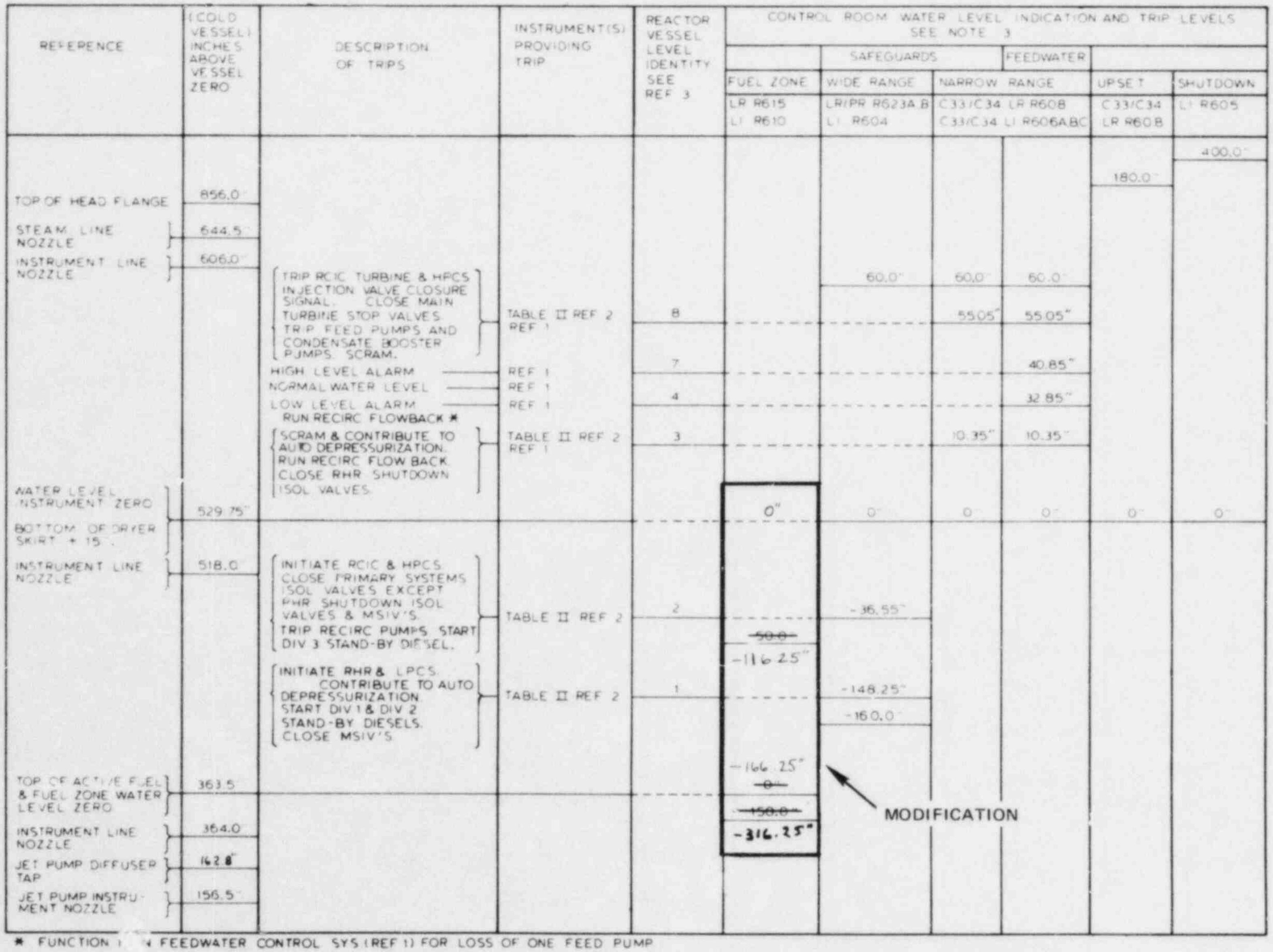

Figure 1A.67-1. Planned Modification for Common Water Level Reference (Figure 5.2-11)

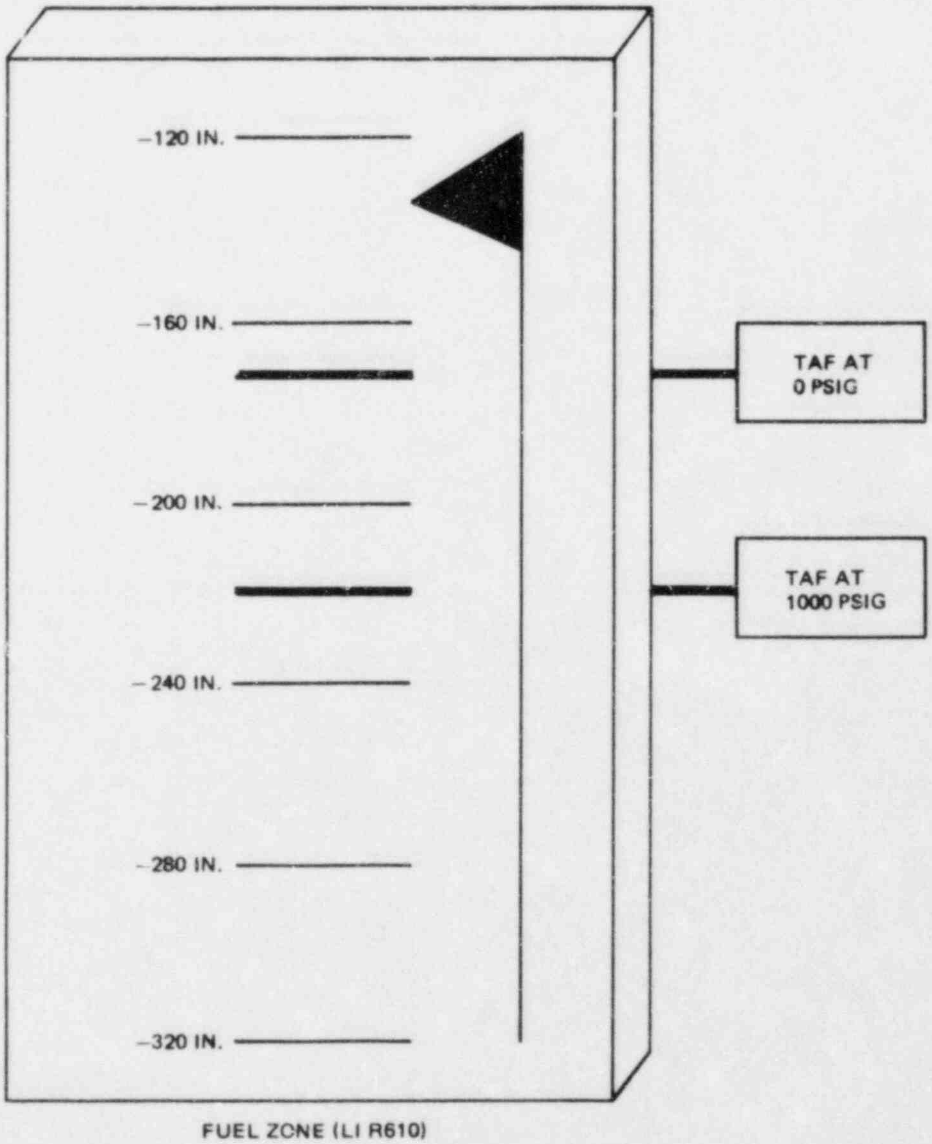

Figure 1A.67-2. Fuel Zone Common Reference Zero

1A.68 VERIFY QUALIFICATION OF ACCUMULATORS ON AUTOMATIC DEPRESSURIZATION SYSTEM VALVES (NUREG-0737 Item II.K.3.28)

NRC Position

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. GE has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the Licensee must show that the accumulator design is still acceptable.

Response

The accumulators for the ADS valves are sized to provide two operating cycles at 70% of drywell design pressure. This cyclic capability is validated during preoperational testing at the station. The accumulators are safety grade ASME Section III components.

The 100-day, post-accident functional operability requirement is met through conservative design and redundancy; eight ADS valves are provided with code-qualified accumulators and seismic Category I piping within primary containment. Two redundant 7-day supplies of bottled air are available for long-term usage with replacement capability being provided for the remainder of the postulated accident to assure system functional

1A.68 VERIFY QUALIFICATION OF ACCUMULATORS ON AUTOMATIC DEPRESSURIZATION SYSTEM VALVES (NUREG-0737 Item II.K.3.28) (Cont'd)

Response (Cont'd)

operability. A maximum of three ADS valves need function to meet short-term demands and the functional operability of only one ADS valve will fulfill longer term needs. Each accumulator is instrumented to provide the reactor operator with indication of the failure of any of the redundant systems under hostile environmental condition.

The BWR Owners' Group sponsored an evaluation of the adequacy of the ADS configurations. Evaluation results are summarized in the following paragraph.

The accumulators are designed to provide two ADS actuations at 70% of drywell design pressure, which is equivalent to 4 to 5 actuations at atmospheric pressure. The ADS valves are designed to operate at 70% of drywell design pressure because that is the maximum pressure for which rapid reactor depressurization through the ADS valves is required. The greater drywell design pressures are associated only with the short duration primary system blowdown in the drywell immediately following a large pipe rupture for which ADS operation is not required. For large breaks which result in higher drywell pressure, sufficient reactor depressurization occurs due to the break to preclude the need for ADS. One ADS actuation at 70% of drywell design pressure is sufficient to depressurize the reactor and allow inventory makeup by the low pressure ECC systems. However, for conservatism, the accumulators are sized to allow 2 actuations at 70% of drywell design pressure.

See Subsection 6.8.1 for a description of the ADS air supply.

1A.71 EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO FUEL FAILURE (NUREG-0737 Item II.K.3.44)

NRC Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result from a stuck-open relief valve should be included in this category.

Response

The BWR Owners' Group sponsored an evaluation of the worst anticipated transient with the worst single failure. These results were submitted to the NRC via a letter from D. B. Waters, Chairman BWR Owners Group, to D. G. Eisenhut, NRC, dated December 29, 1980. (Reference 20). A letter (Reference 26) from D. G. Eisenhut (NRC) to D. B. Waters (BWR Owner's Group) transmitted the NRC evaluation of this item. The staff found that the report was acceptable for referencing by individual licensee/ applicants. A summary of the results of the analysis follows.

The anticipated transients in NRC Regulatory Guide 1.70, Revision 3 were reviewed for all BWR product lines from the BWR/2 through BWR/6 from a core cooling viewpoint. The loss of feedwater event was identified to be the most limiting transient which would challenge core cooling. The 238 Nuclear Island is designed so that the HPCS, RCIC or ADS with subsequent low pressure

1A.71 EVALUATION OF ANTICIPATED TRANSIENTS WITH SINGLE FAILURE TO VERIFY NO FUEL FAILURE (NUREG-0737 Item II.K.3.44) (Cont'd)

Response (Cont'd)

makeup is each independently capable of maintaining the water level above the top of the active fuel given a loss of feedwater. The detailed analysis in Reference 20 shows that even with the worst single failure in combination with the worst transient the core remains covered.

Furthermore, even with degraded conditions involving one SORV in addition to the worst transient with the worst single failure, these studies show that the core remains covered during the whole course of the transient either due to RCIC operation or due to automatic depressurization via the ADS or manual depressurization by the operator so that low pressure inventory makeup can be used.

It is concluded that for anticipated transients combined with the worst single failure, the core remains covered. Additionally, it is concluded that for severely degraded transients beyond the design basis where it is assumed that a SRV sticks open and an additional failure occurs, the core remains covered with proper operator action.

1A.74 UPGRADE EMERGENCY PREPAREDNESS (NUREG-0737 Item III.A.1.1)

NRC Position*

Comply with Appendix E, "Emergency Facilities" to 10 CFR Part 50, Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," and for the offsite plans, meet essential elements of NUREG-75/111 or have a favorable finding from FEMA.

Response

The response to this requirement will be supplied by the applicant.

Section 1A.8 provides additional information on procedures for transients and accidents.

*This position statement is repeated from Reference 22 since it was not given in detail in NUREG 0737.

1A.80 REFERENCES

1. U. S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC report NUREG-0660, Vols. 1 and 2, May, 1980.

2. U. S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," USNRC Report NUREG-0737, November, 1980.

3. Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees and Applicants, Subject: Interim Criteria for Shift Staffing, dated July 31, 1980.

4. Letter from D. G. Eisenhut, NRC, to All Operating Nuclear Power Plants, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident, dated September 13, 1979.

5. Letter from D. B. Vassallo, NRC, to All Pending Operating License Applicants, Subject: Followup Actions Resulting from the NRC Staff Reviews Regarding the Three Mile Island Unit 2 Accident, dated September 27, 1979.

6. Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, Subject: Emergency Planning, dated October 10, 1979.

7. Letter from H. R. Denton, NRC, to All Operating Nuclear Power Plants, Subject: Discussion of Lessons Learned Short-Term Requirements, dated October 30, 1979.

1A.80 REFERENCES (Cont'd)

8. Letter from D. B. Vassallo, NRC, to All Pending Operating License Applications, Subject: Discussion of Lessons Learned Short-Term Requirements, dated November 9, 1979.

9. Letter from D. G. Eisenhut, NRC, to S. T. Rogers, BWR Owners' Group, Subject: Emergency Procedure Guidelines, dated October 21, 1980.

10. Letter from D. B. Vassallo, NRC, to All Pending Construction Plant Applicants Subject: Discussion of Lessons Learned Short-Term Requirements, dated November 9, 1979.

11. U. S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Final Report," USNRC Report NUREG-0585, October 1979.

12. U. S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," USNRC Report NUREG-0578, July, 1979.

13. U. S. Nuclear Regulatory Commission, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC Report NUREG-0660, Appendix C, Table C.1, item 5.

14. U. S. Nuclear Regulatory Commission, Office of Inspection and Enforcement Region III, "Nuclear Incident at Three Mile Island-Supplement," IE Bulletin No. 79-05B, April 1979.

1A.80 REFERENCES (Cont'd)

15. U. S. Nuclear Regulatory Commission, "TMI-Related Requirements for New Operating Licenses," USNRC Report NUREG-0694, June 1980.

16. Letter from D. F. Ross, NRC, to All B&W Operating Plants (except TMI-1 and -2), Subject: Identification and Resolution of Long-Term Generic Issues Related to the Commission Orders of May 1979, dated August 21, 1979.

17. Reports of the Bulletins and Orders Task Force of the NRC Office of Nuclear Reactor Regulation:

    a. U. S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants," USNRC Report NUREG-0611, January 1980.

    b. U. S. Nuclear Regulatory Commission, "Staff Report of the Generic Assessment of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Boiling Water Reactors Designed by the General Electric Company," USNRC Report NUREG-0626, January 1980.

18. U. S. Nuclear Regulatory Commission, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors," USNRC Report NUREG-0623, November 1979.

1A.80 REFERENCES (Cont'd)

19. Letter from D. B. Waters, Chairman, BWR Owners' Group, to D. G. Eisenhut, NRC, dated March 31, 1981, Subject: BWR Owners' Group Evaluations of NUREG-0737 Requirements II.K.3.16 and II.K.3.18.

20. Letter from D. B. Waters, Chairman, BWR Owners' Group, to USNRC dated December 29, 1980, Subject: BWR Owners Group Evaluation of NUREG-0737 Requirements.

21. Letter from D. B. Waters, Chairman BWR Owners' Group, to USNRC, dated May 22, 1981, Ltr No. BWROG-8142, Subject: BWR Owners' Group Evaluation of NUREG-0737, Item II.K.3.25, "Effect of Loss of Alternating Current Power on Pump Seals."

22. U. S. Nuclear Regulatory Commission (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," USNRC Report NUREG-0654, January 1980.

23. Letter from D. G. Eisenhut, NRC, to All Power Reactor Licensees, Subject: Clarification of NRC Site Requirements for Emergency Response Facilities at Each Site, dated April 25, 1980.

24. Letter from R. H. Buchholz, GE, to D. F. Ross, NRC, Subject: Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, November 30, 1979, MFN-290-79.

1A.80 REFERENCES (Cont'd)

25. Letter to R. H. Vollmer (NRC) from D. B. Waters (BWR Owners Group), "NUREG-0578 Requirement 2.1.2 - Performance Testing of BWR and PWR Relief and Safety Valves", September 17, 1980.

26. Letter from D. G. Eisenhut (NRC) to all Applicants/ Licensees Referencing BWR etc., Subject: NUREG-0737, Item II.K.3.44 - Evaluation of Anticipated Transients Combined with Single Failure (Generic Letter No. 81-32), August 7, 1982.

1AA.2 SUMMARY OF SHIELDING DESIGN REVIEW (Continued)

rooms and pumps and valves per Table 1AA-1. All vital equipment will be environmentally qualified. It is also shown that this exposure envelope is not time dependent after about 100 days.

c) It is not necessary for operating personnel to have access to any place other than the Control Room and three manual valves in the Auxiliary and Fuel Buildings to operate the equipment of interest during the 100 day period. The manual valves are for essential service water supply (one in each division) to the hydrogen mixing blowers of the Combustible Gas Control system and a Drywell Bleed-off Vent System valve. These valves are considered accessible on a controlled exposure basis. Direct shine from the containment is less than one R/hr within four hours post-accident.

d) The control room is designed to be accessible post-accident.

e) Access to radwaste is not required, but the Radwaste Building is accessible since primary containment sump discharges are isolated and secondary containment sump pump power is shed at the onset of the accident. Thus, fission products are not transported to radwaste. The hydrogen control system is operated from the Control Room; the 238 Nuclear Island does not have a containment isolation reset control area or a manual ECCS alignment area. These functions are provided in the Control Room.

1AA.2 SUMMARY OF SHIELDING DESIGN REVIEW (Continued)

f) Within days to a month following an accident, access would be possible to electrical equipment rooms containing motor control centers and corridors in the Auxiliary Building and to various areas in the Fuel Building. This is based on radiation shine from the ECCS rooms and primary containment; there is no airborne radiation source in the electrical equipment rooms and ECCS corridor areas. While not necessary to maintain safe shutdown, such access can be useful in extending system functionality and in plant recovery.

g) The emergency power supplies (diesel generators) are accessible within about 100 hours post-accident per Figure 1AA-1. However, access is not necessary since the equipment is environmentally qualified.

1AA.3 CONTAINMENT DESCRIPTION AND DEFINITION OF TERMS

1AA.3.1 Description of Primary/Secondary Containment

The 238 Nuclear Island includes many features to assure that personnel occupancy is not unduly limited and safety equipment is not degraded by post-accident radiation fields or during other operating periods. While these features are described in other sections, a brief review in the present context is provided here for emphasis.

The configuration of the drywell and primary containment and the suppression pool maximizes the scrubbing action of fission products by the suppression pool. The particulate and halogen content of the primary containment atmosphere following an accident is thereby substantially reduced compared to the Regulatory Guide 1.3 source terms. The calculations for this design review were made prior to General Electric Company's suppression pool scrubbing tests (Subsection 15D.2.2) so these calculations do not reflect the substantial decontamination factors found in the tests. The calculations therefore have extensively higher primary containment (and secondary containment) airborne radioactivity concentrations than would be the real case.

The secondary containment consists of the Shield Building annulus, the Fuel Building and the ECCS rooms of the Auxiliary Building. Primary containment leakage is limited to less than one percent of the primary containment volume per day by the construction and by the Isolation Valve Leakage Control Systems, Subsection 6.5.3.3, which seal penetration isolation valve leak paths. These control systems are unique to the 238 Nuclear Island design. Of this one percent, leakage to the Fuel Building and ECCS portion

1AA.3.1 Description of Primary/Secondary Containment (Continued)

of the Auxiliary Building is less than 8%, i.e., 0.0008 of the primary containment volume per day. Entry of fission products and their accumulation in the Fuel Building is limited. The source term for the airborne level in these areas is thus minimal.

The Standby Gas Treatment System (SGTS) operates automatically from the beginning of an accident to control the secondary containment pressure to (-)1/4" w.g. The Shield Building acts as a mixing chamber to dilute any primary containment leakage before processing by SGTS and discharge to the environment. Discharge of radioactivity is thus controlled and reduced. Radioactivity content of secondary containment atmosphere is reduced with time by SGTS treatment as well as by decay. (However, prior removal of halogens by scrubbing in the suppression pool offsets the degree of this treatment).

Each ECCS pump and supporting equipment is located in an individual shielded, watertight compartment. Spread of radioactivity among compartments is thus limited. Radiation to the corridors and other areas of the Auxiliary Building is limited to shine through the walls; there is no airborne radiation in these other areas. These areas, outside of ECCS rooms, contain plant electrical control equipment, portions of leakage control systems and HVAC systems. When these become accessible (see Table 1AA-7), any component failures can be repaired thereby improving systems availability.

1AA.3.2 Vital Area

A vital area is any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident.

Areas which must be considered as vital after an accident are the Control Room, Technical Support Center, Sampling Station and Sample Analysis area. The vital areas also include consideration (in accord with NUREG-0737, II.B.2) of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area, motor control centers, instrument panels, emergency power supplies, Security Center and radwaste control panels. Other areas specific to the 238 Nuclear Island to be considered are those for the ADS pneumatic air supply and the auxiliary systems necessary for the operation of the ECCS systems, i.e., power, cooling water, and air cooling. Those vital areas which are plant unique as to location, i.e. Technical Support Center and Security Center, are normally areas of mild environment allowing unlimited access and therefore, are not reviewed for access.

1AA.3.3 Post Accident Operation

Post-accident operations are those necessary to 1) maintain the reactor in a safe shutdown condition, 2) maintain adequate core cooling, 3) assure containment integrity and 4) control of radioactive ventilation releases within 10CFR100 guidelines.

Many of the safety related systems are required for reactor protection or to achieve a safe shutdown condition. However, they aren't necessarily needed once a safe shutdown condition is achieved. Thus, the systems considered herein are only the Engineered Safety Features (ESF) (see Chapter 6) used to maintain the plant in a safe shutdown condition.

1AA.3.3 Post Accident Operation (Continued)

For purposes of this review the plant is assumed to remain in the safe shutdown condition.

The basis for this position is that the foundation of plant safety is the provision of sufficient redundancy of systems and logic to assure that the plant is shutdown and that adequate core cooling is maintained. Necessary shutdown and post-accident operations are performed from the Control Room, except for the several manual valves noted earlier.

1AA.4 DESIGN REVIEW BASES

1AA.4.1 Radioactive Source Term

The radioactive source term used is equivalent to the source terms recommended in Regulatory Guides 1.3 and 1.7 and Standard Review Plan 15.6.5 with appropriate decay times. Depressurized coolant is assumed to contain no noble gas. There is no leakage outside of secondary containment other than via SGTS.

Dose rates for areas requiring continuous occupancy may be averaged over 30 days to achieve the desired <15 mRem/hour.

Design dose rates for personnel in a vital area are such that the guidelines of General Design Criteria (GDC) 19 (i.e., <5 Rem whole body or its equivalent to any part of the body) are not exceeded for the duration of the accident, based upon expected occupancy.

1AA.4.2 Safety-Related Equipment Requiring Qualification

The safety-related equipment requiring review for qualification is only that necessary for post-accident operations and for providing information for assuring post-accident control.

1AA.4.3 Qualification Duration of Safety-Related Equipment

In 10CFR50 the long-term cooling capability is given as follows: "... decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core." A 100 day period has been selected as a sufficient extended period for equipment qualification permitting site and facility response to terminate the event.

1AA.4.4 Availability of Off-Site Power

Since the availability of off-site power is not influenced by plant radiation levels, recovery of off-site power can be achieved without consideration of plant accident conditions. Therefore, even though loss of off-site power may be assumed as occurring coincident with the beginning of the accident sequence, continued absence of off-site power for the accident duration is not realistic. While restoration of off-site power is not a necessary condition for maintaining core cooling, its availability can permit operation of other plant sytems which would not otherwise be permitted by emergency power restrictions, e.g. operation of the Pneumatic Air System, non-safety related HVAC systems and other systems useful to plant cleanup and recovery.

Based on Table A.6-2 of Section 15D.3, the probability for off-site power recovery is 0.999 in 15 hours. This is conservative since the longest time for restoration of off-site power was six hours for the Pennsylvania-New Jersey-Maryland interconnection, the grid used as a basis for the probabilistic risk assessment presented in Section 15D.3.

1AA.4.5 Accidents Used as the Basis for the Specified Radioactivity Release

Table 15.0-3 summarizes the various design basis accidents and associated potential for fuel rod failure. This chapter also provides the accident parameters. Of those accidents only the DBA-LOCA may produce 100% failed fuel rods under NRC worst-case assumptions. The rod drop accident and fuel handling accident are the only other accidents postulated as leading to failed fuel rods with the potential consequence of radioactivity releases.

1AA.4.6 Environmental Qualification Conditions (Continued)

Radiation sources in the secondary containment (especially the ECCS rooms of the Auxiliary Building) are the same as the Table 1AA-2 design basis values for water sources. For airborne radiation sources the plant design basis of Table 1AA-2 for air is used except the primary containment leakage rate to the equipment areas of secondary containment (8% of 1%) is used. Conservatively, the entire 0.08% primary containment leakage is assumed to occur in each of the individual secondary containment compartments. This leakage is limited by the Fission Product Control Systems (Subsection 6.5.3.3). As previously noted, no credit has been taken for the radio-halogen scrubbing which is an inherent feature of the BWR.

1AA.5 RESULTS OF THE REVIEW

1AA.5.1 Systems Required Post-Accident

This section establishes the various systems which are required to function following an accident along with their locations. In the following sections the expected environmental conditions and access and control needs are established for the required post accident systems.

1AA.5.1.1 Necessary Post-Accident Functions and Systems

Following an accident and assuming that immediate plant recovery is not possible, the following functions* are necessary:

a) Reactivity control
b) Reactor core cooling
c) Reactor coolant system integrity
d) Primary reactor containment integrity, and
e) Radioactive effluent control

Reactivity control is a short-term function and is achieved when the reactor is shutdown. The remaining functions are achieved in the longer term post-accident period by use of:

a) The Emergency Core Cooling System (ECCS) and their auxiliaries (for reactor core cooling),

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*ANSI/ANS 4.5 Criteria for Accident Monitoring Functions in Light Water Cooled Reactors

1AA.5.1.1 Necessary Post-Accident Functions and Systems (Continued)

b) The Fission Product Removal and Control Systems and auxiliaries (for containment integrity and radioactive effluent control),

c) The Combustible Gas Control System (CGCS) and auxiliaries (for reactor coolant system and primary containment integrity), and

d) Instrumentation and controls associated with the post-accident monitoring and functioning of the above systems and associated Habitability Systems.

1AA.5.1.2 Emergency Core Cooling Systems and Auxiliaries

Table 1AA-3 shows various systems related to cooling the fuel under post-accident conditions. The table has two purposes:

a) to show what major cooling equipment and systems are required to function simultaneously and thereby define the systems for review, and

b) to show the equipment locations.

This table shows for example that a diesel generator, ECCS equipment and equipment coolers in a ECCS room and essential service water in the same division must all perform together to provide an ECCS function.

As indicated by Table 1AA-3 and Subsection 6.3.1.1.2, it is required that any two of the three combinations tabulated under divisions 1, 2, 3 plus ADS are necessary to achieve safe shutdown within the single failure criterion and loss

1AA.5.1.2 Emergency Core Cooling Systems and Auxiliaries (Continued)

of off-site power. Under these assumptions at least one RHR heat exchanger is available for cooling purposes. The cooling function can also satisfy the containment cooling function in that by cooling suppression pool water, which is the source of water flowing to the reactor, the containment source of heat is also removed.

The diesel generator, electrical switchgear and essential service water system of the same divisions, will also be needed. As a minimum, this combination must last until other actions can be taken (e.g., for 100 days). However, see Section 4.4 regarding the needs for diesel generators.

The fuel pool cooling function is also included in Table 1AA-3 on the basis that a recently unloaded fuel batch could require continued cooling during the post-accident period. The FPCCS equipment is environmentally qualified so access is not required and redundancy is included in system components.

The Automatic Depressurization System (ADS) pneumatic air supply is included in the tabulation since a postulated non-break or small break accident could require continued need for the depressurization function until the RHR system is placed in the shutdown reactor cooling mode. In the case of a non-break or a small break accident, the majority of the fission products would be released via the safety relief valves to the suppression pool and hence to the containment rather than direct mixing through the horizontal vents as would occur following a DBA-LOCA. In either case the distribution of fission products is assumed to be the same as for the DBA-LOCA even though realistically a significant portion

1AA.5.1.2 Emergency Core Cooling Systems and Auxiliaries (Continued)

of halogens and solid fission products would be retained in the reactor pressure vessel. Thus, the results as they apply to the ADS are very conservative.

Table 1AA-4 supplements Table 1AA-3 by showing the location of selected associated valves and instrument transmitters. These do not represent all of this type of equipment which is environmentally qualified, safety-related, or included in the systems of Tables 13.11-2 through 13.11-7. It does however, represent components which are needed to operate, generally during the beginning of the accident. For example, most ECCS system valves are normally open, and only a pump discharge valve needs to open to direct water to the reactor. Similarly, the instrument transmitters shown are those which would provide information on long term system performance post-accident. Control Room instrumentation is not listed since it is all in an accessible area where no irradiation degradation would be expected. Passive elements such as thermocouples and flow sensors are not listed although they are environmentally qualified. The components listed under B21-Main Steam are those for ADS function or monitoring reactor vessel level. Suppression pool level is included with the HPCS instrumentation.

1AA.5.1.3 Fission Product Removal and Control Systems and Auxiliaries

The systems and equipment of interest are shown in Table 1AA-5 and are described in Section 6.5 (except as noted). Included are:

1AA.5.2.2 Discussion of Conservatism in Calculations (Continued)

The electrical equipment rooms in the Auxiliary Building are outside the secondary containment barrier and so are not exposed to airborne radiation. They are located two floors above the HPCS or LPCS rooms so dose rates calculated at three feet above the HPCS or LPCS cover slabs are greater than expected at the equipment rooms and the electrical equipment exposures have been reduced accordingly. Although access is not required during the post-accident period, access is possible for short times after 5 days (2 R/hr) and for progressively longer times after 30 days (400 mR/hr) and 100 days (70 mR/hr).

The above discussion shows that the calculated exposures are high compared to what would actually be the case. The calculations would envelope the actual expected conditions but still be within the environmental requirements. As shown later, the calculated conditions are within the environmental qualification conditions.

1AA.5.2.3 Discussion of Fuel Building Access Needs

Dose rates and exposures in the Fuel Building are of post-accident interest because the following are located there:

a) SGTS equipment
b) ADS air bottles and air compressors and receivers
c) Fuel Pool Cooling and Cleanup Systems
d) The isolation valve leakage control system mechanical equipment

1AA5.2.3 Discussion of Fuel Building Access Needs (Continued)

Equipment which is qualified to the environmental requirements will operate through the post-accident period so access is not required for that reason. While the Water Positive Leakage Control System water supply is designed for a minimum of 30 days without make-up, additional water can be supplied by the Essential Service Water system. The safety related air compressors for the isolation valve leakage control systems are redundant. Thus, these systems will continue to provide isolation valve seal leakage control.

The air supply to the Automatic Depressurization System (ADS) is adequate for seven days in which case replacement of the air bottles could be necessary. In the event of a DBA-LOCA this would not be required. In the event of a non-break or small break LOCA it may be necessary, but in that case the probability of a concurrent fission product release as postulated for the DBA-LOCA is small. Thus, access to the Fuel Building for bottle replacement is expected to be possible. On the other hand restoration of off-site power is highly probable within 15 hours after the accident. This would permit operation of the non-safety grade pneumatic air system compressors which also supply air to the ADS. These compressor systems are redundant and are in a mild environment; thus access to replace the ADS air bottles is not considered necessary under post-LOCA conditions.

1AA.6 CONCLUSIONS OF THE REVIEW

Table 1AA-7 shows that the dose rates in the ECCS rooms will not permit occupancy during the post-accident period of 100 days. However, since the equipment in these rooms will be qualified for operation throughout this period, occupancy is not expected, nor required.

At 7 R/hr after 30 days and 2 R/hr after 100 days the ECCS corridor could be visited for a short time; however, there is no need since equipment located there will be environmentally qualified. Only corridor locations adjacent to operating ECCS rooms will have this conservative dose rate.

At 400 mR/hr after 30 days the electrical equipment rooms can be considered accessible; this drops to 70 mR/hr at 100 days. However, access is not required for continued ECCS operation. Again, only equipment above operating ECCS rooms will have these conservative dose rates. Auxiliary Building self-contained A/C units are environmentally qualified and will provide cooling air to the electrical equipment areas.

The Fuel Building radiation results primarily from airborne radiation due to assumed leakage from the primary containment except for the SGTS and FPCCS areas. In these areas the exposure is principally due to the radioactive material collected on the SGTS filters and circulating in the FPCCS. At 100 days the general Fuel Building dose rate is 20 mR/hr, so at some time after 30 days access is possible for workable periods within the NUREG-0737, II.B.2 exposure guidelines.

Figure 1AA-1 shows the dose rates and integrated dose in the Diesel Generator Building. The Diesel Generator Building is accessible within about 100 hours. However, the equipment is environmentally qualified so access is not required.

1AA.6 CONCLUSIONS OF THE REVIEW (Continued)

From an equipment viewpoint, Figure 1AA-2 shows the dose rate and integrated exposure for the ECCS rooms as a function of time. Three curves of integrated dose are shown for three conditions: 1) average piping dose is derived from the dose rate curve of the figure, 2) the integrated dose curve is based on the peak piping dose rate and gives the highest calculated exposure, 3) the third curve assumes no water leakage and reflects the fact that an operating pump room cannot also have significant leakage over a 100-day operating period because it would be flooded. All of the exposure curves show less integrated dose than the qualification exposure. Therefore, equipment qualified to the requirements would operate through the specified post-accident period.

Since the dose rate declines with time, the calculated integrated dose is always below the required dose regardless of time and does not change appreciably after 100 days. Thus, exposure would not be a factor limiting equipment performance.

Table 1AA-9 compares post-accident and equipment qualification exposures in the areas of interest. The equipment which is qualified to the indicated values will operate through the specified post-accident periods.

Since access is not required for equipment qualified to the environmental requirements, access is not required except in the control room. No changes are therefore necessary. This review has shown that the requirements of NUREG-0737, II.B.2 are satisfied.

TABLE 1AA-1

REQUIRED** ACCIDENT ENVIRONMENTAL CONDITIONS

Location*	Equipment	Integrated Radiation, Rads	
		Beta	Gamma
CT-1 through CT-5	Some instrument transmitters	3.3×10^8	2.2×10^7
AB-1	Elect. Switchgear Remote Shutdown Panel Rooms	$<1 \times 10^3$	1.7×10^5
AB-2	LPCS, HPCS, RHR "C" Rooms	5×10^7	5×10^6 (3)
AB-4	RHR Pump Rooms "A" & "B"	5×10^7	5×10^6 (3)
AB-6	Corridors outside ECCS Rooms	-	1.7×10^5
FB-1	Fuel Pool Pump Areas	4×10^5	2×10^4 (4)
FB-2	Operating Floor	4×10^5	2×10^4
FB-3	Below Operating Floor	4×10^5	2×10^4
CB-1	Control and Control Equipment Rooms	$<5 \times 10^3$	$<1 \times 10^3$
DG Bldg.	Diesel Generator and Auxiliaries incl. HVAC	4.5×10^4	1.0×10^4

*See Table 3.11-1 for locations

**Required is that accident exposure which, when added to the non-accident exposure, will be used for equipment qualification.

Note 1) Exposure is based upon a condition duration of 100 days of DBA-LOCA.

2) Small break accidents have the same or lesser specified exposures.

3) RHR and LPCS pumps and valves = 1.4×10^7 Rads, gamma.

4) SGTS cubical bar 1.4×10^7 Rads, gamma total.

TABLE 1AA-2

RADIATION SOURCE COMPARISON

ACTIVITY GROUP	% CORE INVENTORY RELEASED		
	R.G. 1.3	R.G. 1.7	Plant Design Basis
AIR			
Noble Gases	100	100	100*
Halogens	25	--	25*
All Remaining	--	--	--
WATER			
Noble Gases	0	--	--
Halogens	--	50	50**
All Remaining	--	1	1**

* Uniformly mixed within the primary containment boundary

** Uniformly mixed in the suppression pool and reactor coolant

TABLE 1AA-3

CORE COOLING SYSTEMS AND AUXILIARIES

Division 1				
Location	Equipment		Location	
D.G. Bldg. "A"	Div 1 DG Engine Heat Exchanger*		DG Bldg. "B"	Div
AB-1	Div 1 Elect. Switchgear		AB-1	Div
AB-4 (RHR Room "A")	E12-C002A	RHR Pump "A" (LPCI)	AB-4 (RHR Room "B")	E12
	X73-ECU04	RHR Pump "A" Room Cooler*		X73
	Incl w/pump	RHR Pump "A" Seal Cooler*		Inc
	E12-B001A	RHR Heat Exchanger "A"		E12
AB-2 (LPCS Room)	E21-C001	LPCS Pump	AB-2 (RHR Room "C")	E12
	X73-ECU03	LPCS Pump Room Cooler*		X73
				Inc
FB-3	P53-AA001A	Pneumatic Air Supply Receiver	FB-3	P53
FB-3	P53-AA0012A	Pneumatic Air Supply Air Bottles	FB-3	P53
FB-1	-	Pneumatic Air Supply Non-essential Compressor & Dryer	FB-1	-
FB-1	G41-C001A	FPCCS Pump "A"	FB-1	G41
	X63-ECU02A	FPCCS Pump Room Cooler "A"*		X63
	G41-B001A	FPCCS Heat Exchanger "A"*		G41
CT	G41-F040A	M.O. Valves +	CT	G41
CT	G41-F044A	M.O. Valves	CT	G41
FB-1	G41-N024A	Pressure Transmitter	FB-1	G41

* Equipment cooled by ESW System Div. 1.

** Equipment cooled by ESW System Div. 2.

*** Equipment cooled by HPCS Service Water System

\+ M.O. = Motor Operated

- MAJOR EQUIPMENT

Division 2		Division 3	
Equipment		Location	Equipment
2 DG Engine Heat Exchanger**		HPCS DG Bldg.	HPCS DG HX***
2 Elect. Switchgear			
-C002B	RHR Pump "B" (LPCI)		
ECU07	RHR Pump "B" Room Cooler**		
w/pump	RHR Pump "B" Seal Cooler**		
-B001B	RHR Heat Exchanger "B" **		
-C002C	RHR Pump "C" (LPCI)	AB-2 (HPCS Room)	E22-C001 HPCS Pump
-ECU06	RHR Pump "C" Room Cooler**		X73-ECU08 HPCS Pump Room Cooler***
1. w/pump	RHR Pump "C" Room Cooler**		
-AA001B	Pneumatic Air Supply Receiver		
-AA002B	Pneumatic Air Supply Air Bottles		
	Pneumatic Air Supply		
	Non-essential Compressor & Dryer		
-C001B	FPCCU Pump "B"		
-ECU02B	FPCCU Pump Room Cooler "B"**		
-B001B	FPCCU Heat Exchanger "B"**		
-F040B	M.O. Valves		
-F044B	M.O. Valves		
-N024B	Pressure Transmitter		

TABLE 1AA-4

CORE COOLING SYSTEMS AND AUXILIARIES

VALVES AND INSTRUMENT TRANSMITTERS

System	MPL No.	Equipment Description	Location
B21-Main Steam			
Valves	A003	Air accumulators (for SRV)	DW-1
	A004	Air accumulators (for SRV)	DW-1&2
	F041	Safety releif valves	DW-2
	F047	Safety relief valves	DW-2
	F051	Safety relief valves	DW-2
Pressure Transmitters	N067 C,G,D,H	HPCS	CT-3
	N073 A,B	RHR A,B,C	CT-3
	N094 A,E,B,F	RHR A,B,C; ADS A,B	CT-3
	N097 A,B	RHR A,B,C	CT-3
Level Transmitters	N073 C,G,D,H	RPV level & HPCS	CT-3
	N091 A,B,E,F	RPV level & ADS A,B	CT-3
		RPV level & RHR/LPCI	
	N095 A,B	RPV level & ADS A,B	CT-3
E12 RHR System			
Valves	F008	Reactor to RHR	CT
	F009	Reactor to RHR	DW
	F042	RHR-Hx to reactor	CT-2
Flow Transmitters	N013	SD cooling to RPV head	AB*
	N015	RHR Discharge to reactor	AB*
	N052	RHR Discharge to reactor	AB*
Pressure Transmitters	N053	RHR Pump discharge alarm	AB*
	N055	RHR Pump discharge alarm	AB*
	N056	RHR Pump discharge alarm	AB*
	N057	RPV Recirc. to RHR pump	AB*
	N058 A,B,C	LPCI discharge to RPV	AB*

TABLE 1AA-4

CORE COOLING SYSTEMS AND AUXILIARIES

VALVES AND INSTRUMENT TRANSMITTERS (cont'd)

System	MPL No.	Equipment Description	Location
E21 LPCS			
Valves	F001	Suction valve	AB
	F005	Discharge valve	AB-4
Flow Transmitter	N003	Pump discharge flow	AB-2
Pressure Transmitter	N053	Pump discharge pressure	AB-2
E22 HPCS			
Valves	F001	Cond. storage to pump**	AB-2
	F004	Pump to RPV	AB-2
	F015	Supp. pool to pump**	AB-2
Pressure Transmitters	N051	HPCS Pump discharge press.	AB-2
	N052	HPCS Pump suction press.	AB-2
Flow Transmitter	N005	HPCS Pump discharge flow	AB-2
Level Transmitter	N005 C,G	Suppression pool level	CT
P33 Pneumatic Supply System			
	DD005	Air filter	CT-3
Valves Air Operated	FF015 A,B	Receiver to ADS	FB
	FF017 A,B	Receiver to ADS	CT
	FF037 A,B	Compressor to receiver	FB
	FF038 A,B	Air bottles to receiver	FB
PC Valve***	FF051 A,B	Air bottles to receiver	FB
Press. Switch	NN004 A,B	Air bottle pressure switch	FB

* RHR transmitters are located in the respective RHR rooms

** Needed only to change source of pump suction

*** PC = pressure control

Table 1[illegible]5

FISSION PRODUCT REMOVAL AND CONTROL

System	Equipment	Location			
		Division 1		Division 2	
		MPL	Location Code	MPL	Location Code
P38 SGTS	Exhaust Fan	CC001A	FB-5	CC001B	FB-5
	Heat Removal Fan	CC003A	FB-5	CC003B	FB-5
	SGTS Unit	ZZ001A	FB-4	ZZ001B	FB-4
	System Inlet Valve (AO) (+) (++)	FF001A	FB-4	FF001B	FB-4
	System Inlet Valve (MO)	FF002A	FB-4	FF002B	FB-4
	Air Intake (heat removal)(AO)	FF003A	FB-4	FF003B	FB-4
	Air Intake (heat removal)(MO)	FF004A	FB-4	FF004B	FB-4
	Exhaust Fan Suction (MO)	FF006A	FB-4	FF006B	FB-4
	Heat Removal Fan Suction (MO)	FF007A	FB-4	FF007B	FB-4
	Minimum Flow to Fan (MO)	FF010A	FB-4	FF010B	FB-4
	Charcoal Water Drain (MO)	FF015A	FB-4	FF015B	FB-4
	Charcoal Water Spray (MO)	FF016A	FB-4	FF016B	FB-4
	Decay Heat Removal Damper	FF050A	FB-5	FF050B	FB-5
	Decay Heat Removal Damper	FF051A	FB-5	FF051B	FB-5
	Decay Heat Removal Damper	FF052A	FB-5	FF052B	FB-5
	Decay Heat Removal Damper	FF054A	FB-5	FF054B	FB-5
	System Inlet Temperature	NN603A	FB-4	NN603B	FB-4
	System Differential Pressures	RR012A	FB-4	RR012B	FB-4
	System Differential Pressures	RR013A	FB-4	RR013B	FB-4
	System Differential Pressures	NN019A	FB-4	NN019B	FB-4
	Flow Through System, Flow Transmitter	NN02A	FB-4	NN02B	FB-4
X63 Fuel Bldg. HVAC					
	SGTS room cooling unit*	EUC01A	FB-5	EUC01B	FB-5

TABLE 1AA-5 (Continued)

FISSION PRODUCT REMOVAL AND CONTROL

System	Equipment	Location: Division 1 MPL	Division 1 Location Code	Location: Division 2 MPL	Division 2 Location Code
E33 Main Steam Positive Leakage Control Systems	Inboard System Inlet Valves (MO)++	F007	AB-7		
		F008	AB-7		
	Inboard System Pressure Control	F002	AB		
	Inboard System Injection Valves (MO)	F005	AB		
		F007	AB		
		F008	AB		
	Outboard System Inlet Valves (MO)			F027	AB-7
				F028	AB-7
	Outboard System Pressure Control			F022	AB
	Outboard System Injection Valves (MO)			F025	AB
				F027	AB
				F028	AB
	Pressure Transmitter (Inboard)	N001	AB-6		
	Pressure Transmitter (Inboard)	N002	CT-3		
	Pressure Transmitter (Inboard)	N003	CT-3		
	Pressure Transmitter (Inboard)	N004	AB-6		
	Pressure Transmitter (Inboard)	N005	AB-6		
	Flow Transmitter (Inboard)	N007	AB-6		
	Pressure Transmitter (Outboard)			N021	AB-6
	Pressure Transmitter (Outboard)			N022	CT-3
	Pressure Transmitter (Outboard)			N023	CT-3
	Pressure Transmitter (Outboard)			N024	AB-6
	Pressure Transmitter (Outboard)			N025	AB-6
	Flow Transmitter (Outboard)			N027	AB-6
	Control Panel	H22-P074	AB-6	H22-P073	AB-6

* Cooled by ESW

\+ Air operated

++ Motor operated

TABLE 1AA-5 (Continued)

FISSION PRODUCT REMOVAL AND CONTROL

		Location			
		Division 1		Division 2	
System	Equipment	MPL	Location Code	MPL	Location Code
P60 Water Positive Seal Isolation Leakage Control System	Water Supply Tank	AA-011	FB-3	Dryer/Sep. Stg. Pool	
	Water Supply Valves (MO)	FF034	FB	FF002	FB
		FF035	AB	FF003	AB
		FF036	AB	FF004	AB
		FF049	AB	FF005	AB
		FF056	AB	FF020	AB
		FF057	FB	FF055	FB
	ESW--Tank Fill	FF026	FB-3		
	Condensate--Tank Fill	FF027	FB-3		
	Pressurizing Air To Tank (PVC)	PCV-031	FB-3		
	Tank Level Transmitter	NN004	FB-3		
	Tank Pressure Transmitter	NN003	FB-3		
P61 Air Positive Seal Isolation Leakage Control System	Compressor Package* (non-essential)	CC001A	FB-3	CC001B	AB-9
	Air Receiver	AA001A	FB-3	AA001B	AB-9
	Pressure Control (PVC)	FF002	AB	FF029	AB
	Air Supply Valves (MO)	FF004	AB	FF010	AB
		FF005	AB	FF011	AB
		FF006	AB	FF058	AB
X73 Aux. Bldg. HVAC	Elect. Area A/C Unit	ACU02A	AB-1	ACU02B	AB-1
	Self-Contained A/C Unit*	ACU03	AB-1	ACU04	AB-1

*Cooled by ESW

AO = Air operated
MO = Motor operated
TT = Temperature transmitter
dPT = Differential pressure transmitter
FT = Flow transmitter

TABLE 1AA-6

COMBUSTIBLE GAS CONTROL

System	Equipment	Location: Division 1 MPL	Division 1 Location Code	Division 2 MPL	Division 2 Location Code
T41 Reactor Bldg. HVAC	Shield Annulus Exh./Rec. Fan	CC004A	FB-6	CC004B	FB-6
	Hydrogen Mixing Blower	CC008A	CT-4	CC008B	CT-4
	Drywell Bleed-Off Vent System Manual Valve	FF051	FB*	--	--
	Motor Operated Valve	FF038	FB	--	--
P41 Essential Service Water	Manual Water Valves to H_2 Mixing Blower After Cooler and Oil Cooler	--	--	FF125 FF128	FB**
	Motor Operated Valves	FF170	FB	FF114	AB
T49 Flammability Control System	Thermal Recombiner	Z001	CT-4	--	
	Power Supply Panel	Z001	AB-6	--	
	Control Panel	Z001	CB-1	--	
	Hydrogen Recombiner	ZZ001A	CT-4	ZZ001B	CT-4
X63 Fuel Building HVAC	Room Cooling Units for Shield Building Exhaust Fan***	ECU03A	FB-6	ECU03B	FB-6

* Manual valve is backup for MO-FF038. Note on P&ID says operation after accident should be by person wearing an air pack.

** Access to valves may also require an air pack.

*** Cooled by ESW.

TABLE 1AA-7

COMPARTMENT RADIATION DOSE RATE
VS TIME AFTER ACCIDENT

AREA EVALUATED	RADIATION DOSE RATE, R/HR VERSUS TIME				
	4 hrs	1 day	5days	30 days	100 days
SECONDARY CONTAINMENT					
RHR Rooms					
● AIRBORNE	$5x10^3$	$6x10^3$	$2x10^3$	$4x10^2$	$5x10^1$
● WATERBORNE	$7x10^4$	$2x10^4$	$7x10^3$	$8x10^2$	$3x10^2$
FUEL BUILDING					
● AIRBORNE	200	100	45	3	0.02
● WATERBORNE	0	0	0	0	0
● CONTAINMENT SHINE+	0.5	0.02	0.008	0.002	<0.001
OUTSIDE SECONDARY CONT.					
ECCS Corridor					
● AIRBORNE*	50	60	8	2	0.4
● WATERBORNE*	300	60	20	5	2
ELECT. EQUIP. ROOM					
● AIRBORNE*	0	0	0	0	0
● WATERBORNE*	8	5	2	0.4	0.07

ASSUMPTIONS:

- R.G. 1.3 and 1.7 Source Terms
- 8% of secondary containment airborne leakage in each ECCS compartment.
- 5 GPM liquid leak for 5 days for ECCS compartments(1)
- 10% of liquid leaked becomes airborne in ECCS rooms
- 1 AC/(2)day auxiliary building, 2/3 AC/(2)day fuel building
- No cleanup system in operation

(1) At 5 days, the flooding level is reached with consequent isolation.

(2) AC=air changes

* These sources are the sources within the ECCS rooms contributing to radiation in the outside locations. There is no airborne contamination outside secondary containment.

+Containment shine data also applies to areas outside secondary containment

TABLE 1AA-8

POST ACCIDENT RADIATION EXPOSURE, RADS

Location	Airborne Source	Piping* Source	Fluid Leakage Source	Containment Shine	Total
			Auxiliary Building		
HPCS Room	1.2×10^{6}	2.8×10^{6}	8.6×10^{5}	22	5.0×10^{6}
RHR A & B Room	1.2×10^{6}	2.6×10^{6}	8.6×10^{5}	22	4.7×10^{6}
RHR Room C	1.2×10^{6}	1.6×10^{6}	8.6×10^{5}	22	3.7×10^{6}
LPCS Room	1.2×10^{6}	2.3×10^{6}	8.6×10^{5}	22	4.4×10^{6}
ECCS Corridor**	7×10^{3}	1.4×10^{4}	8.8×10^{3}	22	3.0×10^{4}
Above HPCS Room** EL(-)6'-10"	2.5×10^{4}	9.2×10^{3}	5.1×10^{4}	22	8.6×10^{4}
Above LPCS Room** EL(-)6'-10"	2.5×10^{4}	1.5×10^{4}	5.1×10^{4}	22	9.1×10^{4}
			Fuel Building		
Operating Floor	1.8×10^{4}			22	1.8×10^{4}
FPCC Pump Room	1.8×10^{4}			22	1.8×10^{4}
Below Operating Floor	1.8×10^{4}			22	1.8×10^{4}
SGTS Room	1.4×10^{7}			22	1.4×10^{7}
			Containment		
CT 1 through 5					2.2×10^{7}

* Piping source is the average rather than the maximum for the compartment. Maximum value was used in exposure calculation.

** All radiation sources are within HPCS, RHR and LPCS rooms; calculations were based on limiting room rather than making specific calculations for each room.

TABLE 1AA-9

COMPARISON OF CALCULATED EXPOSURES
VS REQUIRED EXPOSURES
FOR POST ACCIDENT SYSTEMS

Location/Zone	Equipment	Post Accident Exposure, Rads	Required Accident Exposure, Rads
Diesel Gen. Bldg.	Diesel Generator	4.5×10^3	1×10^4
Auxiliary Building			
HPCS Room AB-2	HPCS Pump, Motor, Room Cooler, Valves, Inst., Actuators	5.0×10^6	5×10^6
RHR A & B Room AB-4	RHR Pump, Motor, Hx, Seal Cooler, Room Cooler, Valves, Inst., Actuators	4.7×10^6	5×10^6
RHR Room C AB-2	RHR Pump, Motor, Room Cooler, Valves, Inst., Actuators	3.7×10^6	5×10^6
LPCS Room AB-2	LPCS Pump, Motor, Room Cooler, Valves, Inst., Actuators	4.4×10^6	5×10^6
ECCS Corridor AB-6		3.0×10^4	1.7×10^5
Above HPCS Room El (-)6'-10"		8.6×10^4	—
Zone AB-1	Elect. Switchgear		1.7×10^5
Above LPCS Room El (-)6'-10"		9.1×10^4	—
Zone AB-1	Elect. Switchgear		1.7×10^5
Fuel Building			
Operating Floor		1.8×10^4	2×10^4
FPCC Pump Room		1.8×10^4	2×10^4
Below Oper. Floor		1.8×10^4	2×10^4
SGTS Room		1.4×10^7	2×10^7
Control Room			
Control and Control Equipment Rooms CB-1	Control Instrumentation	$< 1 \times 10^3$	$< 1 \times 10^3$
Containment			
CT-1 through CT-5	Instrument Transmitters	2.2×10^7	2.2×10^7

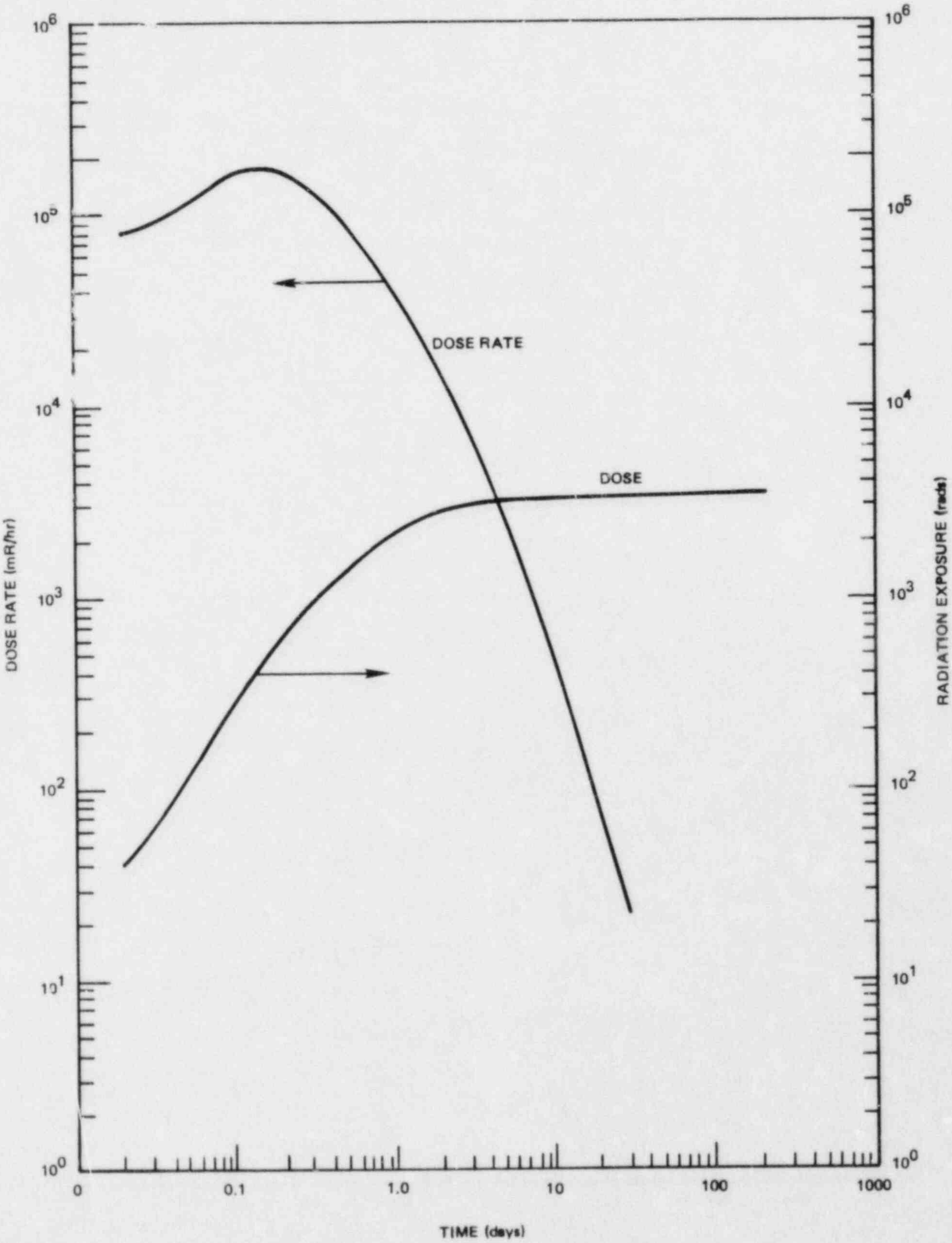
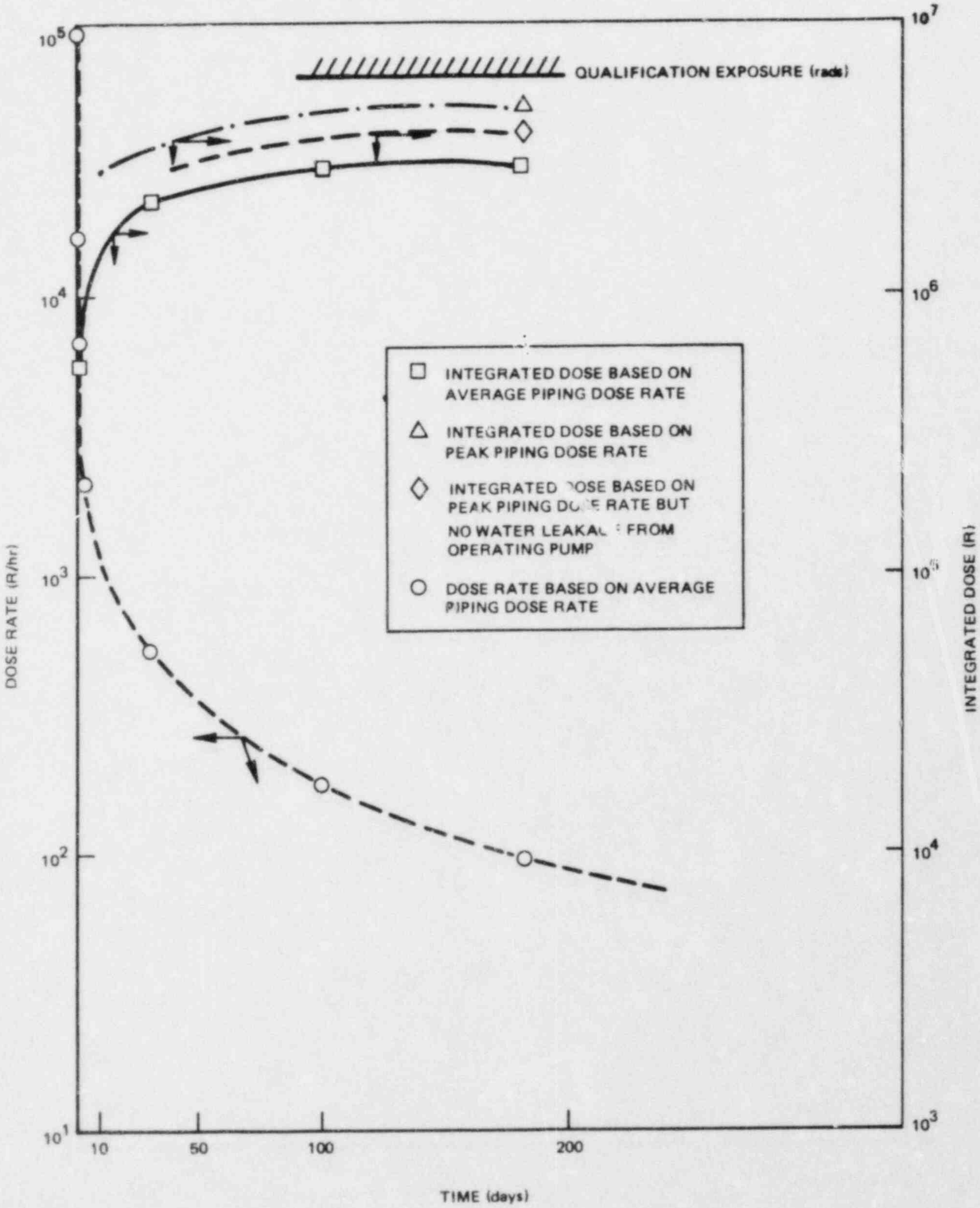


Figure 1AA-1. Diesel Generator Building 1 Radiation Versus Time Following LOCA



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]

Figure 1AA-2. ECCS Rooms - Dose Rate and Integrated Exposure, Post LOCA

1AB.2 GAS SAMPLES

Provision has been made to obtain gas samples from both the drywell and wetwell atmospheres and from the secondary containment atmosphere. The sample system is designed to operate over the range of potential pressures starting at one hour after a LOCA. Heat traced sample lines are used to prevent precipitation of moisture and resultant loss of iodine in the sample lines. The gas samples may be passed through a particulate filter and silver zeolite cartridge for determination of particulate activity and total iodine activity by subsequent counting of the samples on a gamma spectrometer system.

Alternately, the sample flow can bypass the iodine sampler and be chilled to remove moisture. A 15 milliliter grab sample can then be taken for determination of gaseous activity and for gas composition by gas chromatography. This size sample has been adopted to be consistent with present off-gas sample vial counting factors. Provision will be made in the laboratory to aliquot fractions of the initial vial contents to other vials if the activity is too high to count directly.

1AB.3 LIQUID SAMPLES (Cont'd)

includes sample coolers and control valves which select liquid sample points. The station consists of a wall mounted frame and enclosures. Included within the sample station are equipment trays which contain modularized liquid and gas samplers. Each of these modules is approximately 18" x 14" x 20" high. The lower liquid sample portion of the sample station is shielded with 6 inches of lead brick, whereas the upper gas sampler requires 2 inches of lead. The total weight of the wall mounted portion of the system is approximately 7000 pounds. The dimensions of the sample station including shielding is approximately 29" wide by 27" deep by 72" high. The frame is mounted so that the bottom of the frame is approximately 20 inches off the floor. The control instrumentation is installed in a 2' by 4' by 6' high standard cabinet control panel. The panel contains the conductivity, radiation level readouts, and the flow, pressure, and temperature indicators, and various control valves and switches. The general front panel arrangement is shown in Figure 1AB.3-2.

Appropriate sample handling tools are included with the basic sample station. A gas sampler vial positioner and gas vial cask is included. The gas vial is installed and removed by use of the vial positioner through the front of the gas sampler. The vial is then manually dropped into the cask with the positioner which allows the vial to be maintained about 3 feet from the individual performing the operation.

The small volume liquid sample is remotely obtained through the bottom of the sample station by use of the small volume cask and cask positioner. The cask positioner holds the cask and positions the cask directly under the liquid sampler.

1AB.3 LIQUID SAMPLES (Cont'd)

The sample vial is manually raised within the cask to engage the hypodermic needles. When the sample vial has been filled, the bottle is manually withdrawn into the cask. The sample vial is always contained within lead shielding during this operation. The cask is then lowered and sealed prior to transport to the laboratory.

A large volume cask and cask positioner is used to remotely obtain the large volume liquid sample. The positioner contains the cask and vial.

The cask is transported to the required position under the sample station by a four wheel dolly cask positioner. When in position, this cask is hydraulically elevated approximately 1.5 inches by a small hand pump for contact with the sample station shielding under the liquid sample enclosure. The sample bottle is raised, held, and lowered by a simple push/pull cable. The cask is sealed by a threaded top plug inserted above the sample bottle. The weight of this large volume cask is approximately 700 pounds.

The cask may be used for offsite shipment of the large volume sample; however, it will require additional packaging.

A 15 milliliter bottle is contained within the lead shielded cask. This sample bottle is raised from its location in the cask to the sample station needles for bottle filling. The sample station will only deliver 10 milliliters to this sample bottle. When filled, the bottle is withdrawn into the cask. The sample bottle is always shielded by 5-6 inches of lead when in position under the sample station and during the fill and withdraw cycles, thus operator exposure is controlled.

1AC.2 238 NUCLEAR ISLAND ASSESSMENT

The 238 Nuclear Island design has been reviewed against the seven position items of NUREG-0737, Item II.E.4.2. The conclusions of that review follow:

(1) SRP 6.2.4 Compliance

Isolation provisions described in Table 6.2.4 were reviewed and found to meet the recommendations of NRC Standard Review Plan 6.2.4 (Rev. 1).

(2) Essential vs. Non-Essential Classification

The classification of essential and non-essential BWR systems has been defined in NEDO-24782. The results of that classification were used as the basis for the classification of systems in the 238 Nuclear Island design, provided in Table 1AC.2-1.

(3) Non-Essential System Isolation

Non-essential systems are isolated by the containment isolation signals, and by redundant safety grade isolation valves.

(4) Isolation Seal-In Logic

Resetting isolation logic should not cause automatic reopening of containment isolation valves. Eight NSSS valves do not meet this criteria. Design changes to ensure compliance with the criteria are described in Subsection 1AC.3 of this Attachment.

1AC.2 238 NUCLEAR ISLAND ASSESSMENT (Cont'd)

(5) Minimum Containment Pressure Setpoint

The containment pressure setpoint has been reviewed by General Electric and the BWR Owners' Group and was found to be satisfactory.

The results of that evaluation are described in a letter from D. B. Waters, Chairman BWR Owners' Group, to D. G. Eisenhut, (NRC), dated December 29, 1980 (Reference 20).

(6) Sealed Isolation Design

Isolation design provides for a sealed isolation function as discussed under Item (4), above. The normal operation purge lines meet the criteria of this position as discussed in Subsection 1AC.4 to this attachment. The 42" high purge supply and exhaust lines are isolated with a blind flange during normal operation.

(7) Supply and Exhaust and Vent High Radiation Isolation

238 Nuclear Island supply and exhaust isolation valves are provided with a high radiation isolation signal. This signal is provided with an administrative-controlled manual override to mitigate a fuel handling accident or to provide access to the Mark III containment.

1AC.3 MODIFICATIONS TO MEET POSITION ITEM 4

Position Item 4 states: "The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves requires deliberate operator action".

Clarification 4 to NUREG-0737, Item II.E.4.2, further states: Administrative provisions to close all isolation valves manually before resetting the isolation signals is not an acceptable method of meeting this position.

The specific valves of the 238 Nuclear Island containment isolation system whose previously defined logic will allow automatic isolation valve reopening, if the operator acts to RESET the system's isolation valve control logic, are identified below.

	SYSTEM	MPL #
o	RHR Sample Line	E12-F060A/B
		E12-F075A/B
o	Reactor Water Sample Lines	B33-F019
		B33-F020
o	RCIC Steam Supply Lines	E51-F063
		E52-F064

The following subsections described the functional modifications that are planned to the 238 Nuclear Island's present design to meet NUREG-0737, Item II.E.4.2 position Item 4 as previously discussed.

16E12

1AC.3 MODIFICATIONS TO MEET POSITION ITEM 4 (Cont'd)

1AC.3.1 RHR and Reactor Water Sample Lines

For these systems, the modification to the present design is to replace the existing two-position maintained contact switch for each valve with a three-position switch (momentary contact, spring return to "NORMAL" from both the "CLOSE" and "OPEN" mode positions) and to add two new relays for each valve circuit (Figure 1AC.3-1). These modifications in no way curtail the automatic primary isolation function initiation caused by the trip logic function. However, this modification does prevent the isolation valves from opening simultaneously as a consequence of resetting the trip logic function. Replacing the maintained contact switch with the new two-stage momentary contact switch assures the operator that, after initiating the trip logic reset function, the isolation valves will remain closed until deliberate control action is taken to individually open each valve. The added relays are utilized to assure separation of both the "OPEN" and "CLOSE" valve control mode functions.

1AC.3.2 RCIC Steam Supply Lines

For the RCIC system, the modifications to the present design adds another stage of contacts (7-8) to the existing single stage, two-position key lock maintained contact switch for each valve circuit (Figure 1AC.3-2). In the present design, when the trip logic function is initiated by depressing the "RESET" push button, after all isolation signals have cleared, the isolation valves open simultaneously; there is no separation of the operator action to RESET from the operator action to begin opening the isolation valves.

1AC.3.2 RCIC STEAM SUPPLY LINER (Cont'd)]

This additional stage to the control switch prevents the isolation valves from opening simultaneously as a consequence of resetting the trip logic function. By adding another stage to the control switch with the same type of configuration (i.e., both stages currently close their contacts in the "OPEN" position switch mode), separation of the trip logic reset function from the deliberate operator action of opening the closed isolation valves is assured after an isolation has been cleared.

Since the control switch must be in the "OPEN" position mode for normal RCIC operation, the added stage of closed contacts provided by the proposed design change effectively blocks the RESET function until the isolation valves are closed. To open the isolation valve(s), the operator must get the key for the control switch, reposition the switch to the "CLOSE" mode and depress the "RESET" pushbutton; this action resets the isolation valve trip logic so that the trip logic function is armed and ready to respond to another system isolation, if required. But only when the operator returns the switch again to the "OPEN" position mode, can the isolation valve be opened under operator manual control.

The design changes described above for the RHR Sample Line, Reactor Water Sample Lines and RCIC Steam Supply Lines provide the deliberate and separate operator actions required by NUREG-0737, Item II.E.4.2 position Item 4.

16I5

Table 1AC.2-1

ESSENTIAL/NONESSENTIAL EQUIPMENT

	System	Essential	Comments
1.	Reactor Head Cooling	No	Not a safety system.
2.	Standby Liquid Control	Yes	Should be available as back-up to CRD system.
3.	Low Pressure Coolant Injection	Yes	Safety system.
4.	Separate Suppression Pool Cooling	Yes	Main heat sink during isolation.
5.	Core Spray (High-Low Pressure)	Yes	Safety systems.
6.	Closed Cooling Water	No	Used for normal operation only. Not required for DBA but is necessary for the recirc, cleanup system operation, and fuel pool heat exchangers.
7.	Containment Atmospheric Control	Yes	Combustible gas control function necessary to eliminate hydrogen/oxygen combustible atmosphere.

16J1

Table 1AC.2-1

ESSENTIAL/NONESSENTIAL EQUIPMENT
(Continued)

	System	Essential	Comments
8.	Containment Spray Cooling	Yes	Necessary to control drywell/containment pressure.
9.	Automatic Depressurization System	Yes	Safety system; control of RPV pressure.
10.	Standby Gas Treatment	Yes	Necessary to control emissions to environment.
11.	Auxiliary Building/Fuel Building Emergency Cooling	Yes	Necessary to cool safety system pumps and motors.
12.	Reactor Core Isolation Cooling	Yes	Necessary for core cooldown following isolation from the turbine condenser and feedwater makeup.
13.	Auxiliary Building/Fuel Building Equipment Drain	Yes/No	If drain is required, the equipment is probably out-of-service; check for independent isolation; drain should not back up and flood essential equipment.

Table 1AC.2-1

ESSENTIAL/NONESSENTIAL EQUIPMENT
(Continued)

	System	Essential	Comments
14.	Drywell and Containment Floor Drains	No	Not necessary for core cooldown.
15.	Emergency Service Water System	Yes	Necessary to remove heat following accident. Includes the ultimate heat sink.
16.	Instrument Air	Yes	Regarded as essential because this system supports safety equipment. Back-up accumulators are available for the safety equipment should the system fail.
17.	Service Air	No	Serves no safety or shutdown function.
18.	Main Steam Line	No	Not required for shutdown.
19.	Feedwater Line	No	Not required for shutdown. Portion that is Class I is essential.

Table 1AC.2-1

ESSENTIAL/NONESSENTIAL EQUIPMENT
(Continued)

	System	Essential	Comments
20.	Reactor Water Sample	No	Not required for shutdown, but would be necessary for post-accident assessment. Post-accident sample is a separate issue.
21.	Control Rod Drive Cooling	Yes	No credit taken for reflood, but is desirable.
22.	Reactor Water Cleanup	No	Not required during and immediately following an accident. Necessary in long-term recovery.
23.	Radwaste Collection	No	Not required for shutdown.
24.	Recirculation System	No	Not required for jet pump plants because core can be cooled by natural circulation.
25.	RHR Heat Exchangers	Yes	Main heat sink during isolation.
26.	RHR Shutdown Cooling	No	Not essential but desirable to use if available. Not redundant, but safety grade.

16J4

1618

GEI-E5

Table 1AC.2-1

ESSENTIAL/NONESSENTIAL EQUIPMENT
(Continued)

	System	Essential	Comments
27.	RHR Vessel Head Spray	No	Not safety system.
28.	RHR Containment Spray	Yes	Necessary to control pressure.
29.	RHR - LPCI Function	Yes	Safety function.
30.	RHR - Steam Condensing Function	No	Not required as safety equipment.
31.	Waste Collector and Surge Tank	No	Not required for shutdown.
32.	Drywell Cooling	No	Used only in normal operation. Desirable to keep running.
33.	Demineralized Water	No	Not assumed available in ECCS analysis.
34.	Condensate Water	No	Not assumed available in ECCS analysis.

GEIC5

Table 1AC.2-1

ESSENTIAL/NONESSENTIAL EQUIPMENT

(Continued)

	System	Essential	Comments
35.	Fuel Pool Cooling	No	Boiling is acceptable, but make-up is necessary. Heat exchanges cooled by RBCCW system.
36.	Drywell Bleed	Yes	Pressure control vent. Back-up to hydrogen control.
37.	Positive Seal System	Yes	Insure that highly radioactive fluids are confined in the reactor building.
38.	Traversing In-Core Probe (TIP)	No	Not required for reactor shutdown cooling.
39.	Fire Protection	Yes	Availability is essential, as the "accident" may be the result of a fire.
40.	Make-up Water Treatment	No	Serves no purposes during and immediately after accident. Longer-term availability necessary.

APPENDIX 1B

CONTENTS

APPENDIX 1B

CONTENTS
(Continued)

APPENDIX 1B

ASSESSMENT OF UNRESOLVED SAFETY ISSUES

1B.1 SUMMARY

This appendix provides a summary of the relevant investigative programs and measures utilized for addressing the unresolved safety issues applicable to the 238 Nuclear Island. Based on the information provided, it can be concluded that the 238 Nuclear Island can be operated without endangering the health and safety of the public.

1B.1.1 Introduction

The NRC continuously evaluates the safety requirements used in its reviews against new information as it becomes available. As new concerns or safety issues are identified, an assessment is conducted to determine the potential need for any immediate action which may be required to assure safe operation. Depending on the results of the assessment, immediate licensing actions or changes in licensing criteria may or may not be necessary. In any event, further study by the NRC may be deemed appropriate to make judgments as to modification of NRC requirements or implementation of backfitting.

These issues are sometimes referred to as "generic safety issues" because they are related to a particular class or type of nuclear facility rather than a specific plant. The NRC has designated certain of these issues to be "unresolved safety issues" which they perceive as needing a more in-depth technical review or study after the NRC staff has made an initial determination that the safety significance does not prohibit continued operation or require immediate licensing actions.

1B.1.2 Objective

The unresolved safety issues were initially identified in NUREG-0510 (Identification of Unresolved Safety Issues Relating to Nuclear Power Plants", January 1979). These issues are updated quarterly in NUREG 0606 ("Unresolved Safety Issues Summary"). The quarterly update provides current programmatic and schedule information and includes information relative to the implementation status of each issue for which technical resolution is complete.

The overall objective of this appendix is to comply with the Atomic Safety and Licensing Appeal Board decision (ALAB-444) that the Safety Evaluation Report (SER) for each plant should contain an assessment of each significant unresolved generic safety issue. The assessment should include a summary description of relevant investigative programs and the measures devised for dealing with the issues on the subject plant.

1B.1.3 238 Nuclear Island Applicability

The unresolved safety issues outlined in NUREG 0606 include all issues for which technical resolution is not considered complete by the NRC. Several apply only to pressurized water reactors, one applies only to operating nuclear power plants, and one applies only to boiling water reactors with a Mark I Containment. The remaining unresolved safety issues which are applicable to the 238 Nuclear Island are given in Table 1B-1. The number of the generic task in the NRC program addressing each issue is given along with the section in which each issue is discussed.

1B.2.2.3 Industry Activities and Resolution Status (Continued)

In order to protect the 238 Nuclear Island emergency core cooling systems (Section 6.3.2.2.5) against the effects of waterhammer, the ECC systems are provided with jockey pumps. These jockey pumps keep the emergency core cooling system lines full of water up to the motor operated injection valves so that the emergency core cooling system pumps will not start pumping into voided lines. In addition, to ensure that the emergency core cooling system lines remain full, vents have been installed and filling procedures established. Further assurance for filled discharge piping is provided by pressure instrumentation that is used to initiate an alarm that sounds in the main control room if the pressure falls below a predetermined setpoint indicating difficulty maintaining a filled discharge line. Should this occur, or if an instrument becomes inoperable, the required action is identified in the Technical Specification.

To provide additional protection against potential waterhammer events in the 238 Nuclear Island, piping design codes require consideration of impact loads. Approaches used at the design stage include: (1) avoiding rapid valve operation; (2) piping layout to preclude water slugs in steam-filled lines; (3) use of snubbers and pipe hangers; and, (4) use of vents and drains. The use of snubbers and pipe hangers are a by-product of protection from seismic loads, however, their use helps to mitigate the effects of waterhammer events.

In addition, a preoperational vibration and dynamic effects test program will be conducted by the applicant, in conjunction with GE, in accordance with Standard OM-3 of the American Society of Mechanical Engineers for all Class 1, Class 2, Class 3 and other piping systems and piping restraints.

1B.2.2.3 Industry Activities and Resolution Status (Continued)

These tests will provide adequate assurance that the piping restraints have been designed to withstand dynamic effects due to valve closures, pump trips, and other operating modes.

Nonetheless, in the unlikely event that a pipe break did result from a severe waterhammer event, core cooling is assured by the emergency core cooling systems and protection is provided against the dynamic effects of such pipe breaks inside and outside of containment.

In the event that the NRC's activities in Task A-1 identify any potentially significant waterhammer scenarios which have not explicitly been accounted for in the design and operation of the 238 Nuclear Island, corrective measures will be implemented. The task has not identified the need for measures beyond those already implemented.

With respect to Task A-1, it is concluded that the 238 Nuclear Island can be operated without undue risk to the health and safety of the public.

1B.2.3 <u>Reactor Vessel Materials Toughness (Task A-11)</u>

1B.2.3.1 Issue Description

Because the possibility of failure of nuclear reactor pressure vessels (RPV) designed to the ASME Boiler and Pressure Vessel Code is remote, the design of nuclear facilities does not provide specific protection against reactor vessel failure. However, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins.

1B.2.3.1 Issue Description (Continued)

Results from reactor vessel surveillance programs indicate that up to approximately 20 operating PWR's will have beltline materials with marginal toughness, relative to the requirements of Appendices G and H of 10CFR Part 50, after comparatively short periods of operation. The NRC has concluded that for most plants now in the licensing process, current criteria, together with the materials currently employed, are adequate to ensure suitable safety margins for reactor vessels throughout their design lives.

1B.2.3.2 NRC Activities

The principal objective of Task A-11 is to develop safety criteria to allow a more precise assessment of safety margins during normal operation, transients and accident conditions in those older reactor vessels that may have marginal fracture toughness.

1B.2.3.3 Industry Activities and Resolution Status

Based upon evaluation of the 238 Nuclear Island reactor vessel materials toughness, it is concluded that adequate safety margins exist to assure brittle failure is avoided during operating, testing, maintenance, and anticipated transient conditions over the life of the unit. The 238 Nuclear Island complies with all requirements specified in 10CFR50 Appendices G and H.

1B.2.3.3 Industry Activities and Resolution Status (Continued)

The materials of the 238 Nuclear Island reactor vessel meet the fracture toughness requirements of NB-2300 of the ASME Code. Based on these requirements and the fabrication techniques employed by the vessel manufacturer, it is estimated that the total fluence over the design life would result in a final fracture toughness value above the minimum charpy impact requirement of 50 foot-pounds. This means that there is adequate toughness to avoid unstable crack growth from an existing defect at all times during design life. In addition, the surveillance program required by Appendix H of 10CFR Part 50 will afford an opportunity to reevaluate the fracture toughness periodically during a minimum of the first half of the design life.

To assure adequate safety margins, adjustment to the nil ductility transition temperature (NDTT) and the development method for pressure/temperature curves are specified in 10CFR50 Appendices G and H. The amount of adjustment to the operating curves is a function of reference temperature, RT_{NDT} which depends upon the fast neutron (>1 Mev) fluence and copper and phosphorous content in the RPV material. For the 238 Nuclear Island, the copper and phosphorus content of the material is closely controlled. Furthermore, high upper shelf toughness is specified. The fast neutron fluence is low with respect to other reactor types because of the additional moderator (water) in the annulus between the core shroud and the RPV. In addition, thermal shock followed by pressurization is not expected to occur in the 238 Nuclear Island. Furthermore, this conclusion was recently confirmed by the NRC staff in a NRC briefing on pressurized thermal shock on September 15, 1981. Therefore, the reactor pressure vessel material toughness (A-11) issue is not relevant to the 238 Nuclear Island.

1B.2.4.2 NRC Activities (Continued)

laboratory contracts, is underway and a range of methods is being considered and tested for feasibility against a sample of some systems interaction candidates derived from Licensee Event Report evaluations.

1B.2.4.3 Industry Activities and Resolution Status

The licensing requirements and procedures used in the 238 Nuclear Island safety reviews address many different types of systems interaction. Current licensing requirements are founded on the defense-in-depth principle. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems, and protection against events such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires, operator errors, and sabotage.

These design provisions supplemented by the current review procedures of the Standard Review Plan, which require inter-disciplinary reviews and which account, to a large extent, for review of potential systems interactions, provide for an adequately safe situation with respect to such interactions. The quality assurance program which is followed during the design, construction, and operational phases for each plant is expected to provide added assurance against the potential for adverse systems interactions.

The development of systematic ways to identify and evaluate systems interactions may reduce the likelihood of common cause failures which could result in the loss of plant safety functions. However, operational experience with BWR plants that have many features similar to the 238 Nuclear Island indicates that the current review procedures and

1B.2.4.3 Industry Activities and Resolution Status (Continued)

criteria described above, supplemented by post-TMI modifications produce a 238 Nuclear Island design that is reasonably from the effects of potential systems interaction. In addition, the Nuclear Safety Operational Analysis (Chapter 15) and the 238 Nuclear Island Probabilistic Risk Assessment, which considered common cause failures, confirm that such interactions are minor contributors to plant risk which has been shown to be significantly below that reported in WASH-1400.

Applicants are expected to provide for a systematic visual inspection by a multidisciplinary team to review the "as-built" condition of the plant areas where physical interactions could potentially result in adverse effects on safety-grade equipment. Visual inspections of the plant are also expected to be conducted by the applicant to investigate spatially coupled systems interactions that could be initiated by seismic events. Any spatial separations that do not meet established design criteria are to be reported for disposition by analysis and/or hardware modification.

With respect to Task A-17, it is concluded that the 238 Nuclear Island can be operated without endangering the health and safety of the public.

1B.2.5 <u>Safety Relief Valve Hydrodynamic Loads (Task A-39)</u>

1B.2.5.1 Issue Description

All BWR/6 plants are equipped with a number of Safety/Relief Valves (S/RVs) to control primary system pressure transients. The S/RVs are mounted on the main steam lines inside the

1B.2.7.1 Issue Description (Continued)

The NRC concern addressed by this Task Action Plan as it applies to boiling water reactors is primarily focused on the potential for degraded ECCS performance as a result of thermal insulation debris that may be blown from pipes in the drywell and by some means get into the suppression pool during a loss-of-coolant accident causing blockage of the pump suction lines. A second concern is potential vortex formation above the pump suctions and subsequent loss of net positive suction head to the ECCS pumps.

1B.2.7.2 NRC Activities

The NRC is investigating the potential for debris from insulation causing blockage of the ECCS pump strainers. The NRC investigation includes analysis of plant specific designs and the types of insulation used. Also, the NRC had conducted full scale containment emergency sump hydraulic tests at Alden Research Laboratory. The NRC's evaluation of the potential for void formation indicates that there is a much lower level of air ingestion due to vortex formation than previously hypothesized by the NRC. The NRC has also found that up to 2 to 4 percent air void can be accommodated without significantly degrading pumping capacity.

1B.2.7.3 Industry Activities and Resolution Status

With regard to potential blockage of the intake lines, it is very unlikely that insulation would be drawn into the ECCS pump suction lines. Insulation dislodged by a LOCA would primarily tend to collect in the region below the reactor inside the weirwall since this area is large in relation to the size of the annulus between the drywell and the weirwall. The debris in the drywell could only potentially by swept into the suppression pool via the horizontal vents.

1B.2.7.3 Industry Activities and Resolution Status (Continued)

Insulation reaching the suppression pool would tend to either sink to the bottom or float on the surface of the pool. The ECCS suction strainers are sufficiently elevated above the bottom of the pool such that the fluid velocity in the direction of the suction strainers at the bottom of the pool is very low. This minimizes the potential for suction strainer plugging. Additionally, the ECCS suction strainers are twice as large as the size required to assure that adequate net positive suction head (NPSH) is available to the ECCS pumps. Thus, the ECCS suction strainer would have to become more than 50 percent plugged before pump performance would be affected. The design is controlled such that the strainers will not become more than 50 percent plugged.

The second concern, potential vortex formation, is not considered a serious concern for the Mark III containment due to the large depth of the pool and the low approach velocities. The 238 Nuclear Island has a minimum suction submergence for the ECCS systems of over 7 feet.

With respect to Task A-43, it is concluded that the 238 Nuclear Island can be operated without endangering the health and safety of the public.

1B.2.8 Station Blackout (Task A-44)

1B.2.8.1 Issue Description

Electrical power for safety systems at nuclear power plants must be supplied by at least two independent divisions. The systems used to remove decay heat and to cool the reactor core following a reactor shutdown are included among the

1B.2.8.1 Issue Description (Continued)

safety systems that must meet these requirements. Power sources for each electrical division for safety systems include offsite alternating current power connections for normal supply and direct current battery charging, an onsite standby emergency diesel generator for alternating current power supply and direct current battery charging, and a stored energy direct current source (battery).

The unlikely loss of all AC power (that is, loss of AC power from the offsite sources and from the onsite source) is referred to as station blackout. In the event of a staton blackout, the capability to cool the reactor core would be dependent on the timely restoration of AC power or the availability of those systems not requiring AC power. The NRC concern is over the probability and consequences of a station blackout event.

1B.2.8.2 NRC Activities

Task A-44 involves a study of the following elements. First, the NRC through technical assistance contracts is evaluating the expected frequency and duration of offsite power loses at nuclear power plants. Next, an estimation of the reliability and an evaluation of the factors affecting the reliability of onsite emergency AC power supplies will be conducted. The risks to the public posed by station blackout events will then be evaluated. From the above information the NRC plans to assess the effectiveness of safety improvements they perceive may reduce public risk from station blackout events.

1B.2.8.2 NRC Activities (Continued)

The issue of station blackout was considered by the Atomic Safety ad Licensing Appeal Board (ALAB-603) for the St. Lucie No. 2 facility. In addition, in view of the completion schedule for Task A-44 (October, 1982), the Appeal Board recommended that the Commission take expeditious action to accommodate a station blackout event. The commission has reviewed their recommendations and determined that some interim measures should be taken at all facilities while Task A-44 is being conducted. NRC Generic Letter 81-04 requested a review and prompt implementation, as necessary, of emergency procedures and a training program for station blackout events. Consequently, interim emergency procedures and operator training for safe operation of the facility and restoration of alternating current power will be implemented by all operating reactors and by applicants prior to their fuel load date which will supplement the existing set of emergency procedure guidelines.

1B.2.8.3 Industry Activities and Resolution Status

A loss of all offsite alternating current power involves a loss of both the preferred and backup sources of offsite power. The design basis, inspection and testing provisions for the offsite power system will be provided by the applicant in Section 8.2.

The 238 Nuclear Island is provided with redundant power supply systems to provide protection against the loss of offsite power. This includes three AC and four DC onsite power supply divisions.

1B.2.9.2 NRC Activities

The NRC objective is to develop a comprehensive and consistent set of shutdown cooling requirements, including the study of alternative means of shutdown decay heat removal and of diverse systems for this purpose.

The study will consist of a generic system evaluation and will result in recommendations regarding possible design requirements for improvements in existing systems. Also, an alternative decay heat removal method may be considered if it is evaluated to significantly reduce the overall risk to the public.

1B.2.9.3 Industry Activities and Resolution Status

The 238 Nuclear Island is designed with several alternative means for the removal of decay heat. The decay heat is normally rejected via the Power Conversion System. This includes the supply of steam to the main turbine, heat being removed in the main condenser and condensate returned to the vessel by the feedwater system. If the condenser is not available, the safety relief valves operate in either an automatic or manual mode to discharge safety heat to the suppression pool with any of 13 pumps available to makeup the subsequent loss in water inventory and the pool cooling system is operated to transfer this heat to the ultimate heat sink. Under normal shutdown conditions, the residual heat removal (RHR) system is effective in removing decay heat. During abnormal shutdown conditions, the water level in the RPV can be reused to flood the steam lines and decay heat can be removed via a safety/relief valve to the suppression pool and then transferred to the ultimate heat sink by use of the pool cooling system. These decay heat removal and inventory makeup systems are summarized in Sections 15D.2 and are described in detail in Sections 5.4 and 6.3.

1B.2.9.3 Industry Activities and Resolution Status (Continued)

Following the TMI accident, General Electric and the BWR Owners Group performed and documented extensive analyses of feedwater transients and small-break loss-of-coolant accidents to support acceptability of current designs including the BWR/6. A report of these analyses was provided to the NRC in NEDO-24708A Revision 1, dated December, 1980. This report documents that adequate core cooling can be assured by the many diverse inventory maintenance and decay heat removal paths for a wide range of transients and accidents.

The 238 Nuclear Island probabilistic risk assessment results (Section 15D.3) indicate that the loss of long-term decay heat removal is not a dominant event. Consequently, improvements in the decay heat removal function would not significantly reduce the overall risk to the public.

With respect to Task A-45 it is concluded that the 238 Nuclear Island can be operated without endangering the health and safety of the public.

1B.2.10 Safety Implications of Control Systems (Task A-47)

1B.2.10.1 Issue Description

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration.

1B.2.10.3 Industry Activities and Resolution Status (Continued)

A few early operating boiling water reactors have experienced reactor vessel overfill transients with subsequent two-phase or liquid flow through the safety/relief valves. Following these early events, commercial-grade high-level trips (Level 8) have been installed in most BWRs including the 238 Nuclear Island to terminate flow from the appropriate systems. Periodic surveillance testing of these high level trips is required by the Technical Specifications. No overfilling events have occurred since the Level 8 trips were installed. High level trips are also provided for the Reactor Core Isolation Cooling and High Pressure Core Spray systems. In addition, the 238 Nuclear Island has a high level scram that reduces the consequences of an overfill event. Both Nuclear Safety Operational Analyses (Chapter 15) and a PRA (Section 15D.3) have been performed and they provide additional assurance that this issue is not a problem for the 238 Nuclear Island.

With respect to Task A-47, it is concluded that the 238 Nuclear Island can be operated without endangering the health and safety of the public.

1B.2.11 <u>Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (Task A-48)</u>

1B.2.11.1 Issue Description

Postulated reactor accidents which result in a degraded or melted core may result in generation and release to the containment of large quantities of hydrogen. The hydrogen is formed from the reaction of the zirconium fuel cladding with steam at high temperatures and/or by radiolysis of

1B.2.11.1 Issue Description

water. Experience gained from the TMI-2 accident has prompted the NRC to consider additional design provisions for handling larger hydrogen releases than those currently required by the regulations.

1B.2.11.2 NRC Activities

In Task A-48 the NRC will investigate the means to predict the quantity and rate of hydrogen generation during degraded core accidents. In addition, the NRC will examine various means to cope with large releases of hydrogen to the containment such as inerting the containment or controlled burning. The potential effects of proposed hydrogen control measures on safety, including the effects of hydrogen burns on safety-related equipment, will also be investigated.

Because of the potential for significant hydrogen generation as the result of an accident, 10CFR Section 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors," and Criterion 41 of the General Design Criteria, "Containment Atmosphere Cleanup," in Appendix A to 10CFR Part 50, require that systems be provided to control hydrogen in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

The current regulation, 10CFR Section 50.44, requires that the combustible gas control system be capable of handling the hydrogen generated as a result of a design basis loss-of-coolant accident. To provide margin, the assumed hydrogen release is five times the amount calculated in demonstrating compliance with 10CFR Section 50.46 or the amount corresponding to reaction of the cladding to a depth of 0.00023 inch, whichever amount is greater.

3B.7 SUPPRESSION POOL BASEMAT LOADS

In addition to the normal, seismic, deadweight and hydrostatic pressure loadings, that section of the basemat which forms the bottom of the suppression pool also experiences dynamic LOCA loads and oscillatory loads during SRV actuation. The SRV loads are discussed in Attachment A.

The outer half of suppression pool floor will experience a 10-psi bulk-pressure load associated with initial air-bubble formation as discussed in Subsection 3B.6.1.3. This pressure rise above hydrostatic is assumed to increase to 21.8 psi at the drywell wall with the increase from 10 psi to 21.8 psi to be assumed linear and distributed over 50% of the pool width as indicated in Figure 3B-67. This specification is based on the observation that the maximum pressure that the initial bubble can ever have is the maximum drywell pressure during the accident. Data trace no. 1 (Figure 3B-18) indicates that the pressure increase is no greater than 10 psi at a point halfway across the suppression pool. Thus, the specification that the pressure increases linearly between this point and the drywell wall will bound the actual pressure distribution. During the condensation and chugging phases of the postulated LOCA blowdown, the loading on the basemat is the same as that on the containment (Subsections 3B.6.1.9 and 3B.6.1.10).

The containment pressure increases to 3 psi due to drywell air carryover and the long-term pressure and temperature increases (Figure 3B-65). The time history of these pressure transients is shown on Figures 3B-55, 3B-66, and 3B-67.]

SRV oscillating loads are defined in Attachment A. The net loading on the suppression pool linear will reverse during the negative pressure phase of the oscillation and this lifting load on the liner needs to be considered during the design process. Where ground water level is a concern, this pressure is also a consideration in the basemat liner design.

SECTION 4.1

CONTENTS

4.1.3.1 Operation (Continued)

counterbalance steam voids in the top of the core and effect significant power flattening.

These groups of control elements, used for power flattening, experience a somewhat higher duty cycle and neutron exposure than the other rods in the control system.

The reactivity control function requires that all rods be available for either reactor "scram" (prompt shutdown) or reactivity regulation. Because of this, the control elements are mechanically designed to withstand the dynamic forces resulting from a scram. They are connected to bottom-mounted, hydraulically actuated drive mechanisms which allow either axial positioning for reactivity regulation or rapid scram insertion. The design of the rod-to-drive connection permits each blade to be attached or detached from its drive without disturbing the remainder of the control system. The bottom-mounted drives permit the entire control system to be left intact and operable for tests with the reactor vessel open.

4.1.3.2 Description of Control Rods

A description of the control rods is given in Subsection 4.2.2.4.1.

4.1.3.3 Supplementary Reactivity Control

The core control requirements are met by use of the combined effects of the movable control rods, supplementary burnable poison, and variation of reactor coolant flow. Description of the supplementary burnable poison is provided in Sections 4.2 and 4.3.

4.1.4 Analysis Techniques

4.1.4.1 Reactor Internal Components

Computer codes used for the analysis of the internal components are listed as follows:

(1) TASA
(2) DYSEA
(3) HEATER
(4) FAP-71
(5) ANSYS
(6) CLAPS
(7) ASIST

Detail description of these programs are given in the following sections.

4.1.4.1.1 TASA

The TASA program is a two-dimensional and axisymmetric, transient, nonlinear temperature analysis program. An unconditionally stable numerical integration scheme is combined with an iteration procedure to compute temperature distribution within the body subjected to arbitrary time- and temperature-dependent boundary conditions.

This program utilizes the finite element method. Included in the analysis are the three basic forms of heat transfer, conduction, radiation, and convection, as well as internal heat generation. In addition, cooling pipe boundary conditions are also treated. The output includes temperature of all the nodal points for the time instants specified by the user. The program can handle multitransient temperature input.

4.1.4.1.1 TASA (Continued)

A number of heat transfer problems related to the reactor pedestal have been satisfactorily solved using the program.

]

4.1.4.1.2 DYSEA

]

The DYSEA (Dynamic and Seismic Analysis) program is a GE proprietary program developed specifically for seismic and dynamic analysis of RPV and internals/building system. It calculates the dynamic response of linear structural systems by either temporal model superposition or response spectrum method. Fluid-structure interaction effect in the RPV is taken into account by way of hydrodynamic mass.

Program DYSEA was based on program SAPIV with added capability to handle the hydrodynamic mass effect. Structural stiffness and mass matrices are formulated similar to SAPIV. Solution is obtained in time domain by calculating the dynamic response mode by mode. Time integration is performed by using Newmark's β-method. Response spectrum solution is also available as an option.

It has been used extensively in all dynamic and seismic analysis of the RPV and internals/building system.

]

4.1.4.1.3 HEATER

HEATER is a computer program used in the hydraulic design of feedwater spargers and their associated delivery header and piping. The program utilizes test data obtained by GE using full-scale mockups of feedwater spargers combined with a series of models which represent the complex mixing processes obtained in the upper plenum, downcomer, and lower plenum. Mass and energy balances throughout the nuclear steam supply system (NSSS) are modeled in detail.

]

4.1.4.1.3 HEATER (Continued)

The program is used in the hydraulic design of the feedwater spargers for each BWR plant, in the evaluation of design modifications, and the evaluation of unusual operational conditions.

4.1.4.1.4 FAP-71 (Fatigue Analysis Program)

The FAP-71 computer code, or Fatigue Analysis Program, is a stress analysis tool used to aid in performing ASME-III Nuclear Vessel Code structural design calculations. Specifically, FAP-71 is used in determining the primary plus secondary stress range and number of allowable fatigue cycles at points of interest. For structural locations at which the $3S_m$ (P+Q) ASME Code limit is exceeded, the program can perform either (or both) of two elastic-plastic fatigue life evaluations: (1) the method reported in ASME Paper 68-PVP-3, or (2) the present method documented in Paragraph NB-3228.3 of the 1981 Edition of the ASME Section III Nuclear Vessel Code. The program can accommodate up to 25 transient stress states of as many as 20 structural locations.

The program is used in conjunction with several shell analysis programs in determining the fatigue life of BWR mechanical components subject to thermal transients.

4.1.4.1.5 ANSYS

ANSYS is a general-purpose finite element computer program designed to solve a variety of problems in engineering analysis.

The ANSYS program features the following capabilities:

(1) Structural analysis, including static elastic, plastic and creep, dynamic, seismic and dynamic plastic, and large deflection and stability analysis.

4.1.4.1.5 ANSYS (Continued)

(2) One-dimensional fluid flow analysis.

(3) Transient heat transfer analysis including conduction, convection, and radiation with direct input to thermal-stress analyses.

(4) An extensive finite element library, including gaps, friction interfaces, springs, cables (tension only), direct interfaces (compression only), curved elbows, etc. Many of the elements contain complete plastic, creep, and swelling capabilities.

(5) Plotting - Geometry plotting is available for all elements in the ANSYS library, including isometric and perspective views of three-dimensional structures.

(6) Restart Capability - The ANSYS program has restart capability for several analyses types. An option is also available for saving the stiffness matrix once it is calculated for the structure, and using it for other loading conditions.

ANSYS is used extensively in GE/NEBG for elastic and elastic-plastic analysis of the reactor pressure vessel, core support structures, reactor internals, fuel and fuel channel.

4.1.4.1.6 CLAPS

CLAPS is a general-purpose, two-dimensional finite element program used to perform linear and nonlinear structural mechanics analysis. The program solves plane stress, plane strain and axisymmetric problems. It may be used to analyze for instantaneous pressure, temperature and flux changes, rapid transients and steady-state, 4.1.4.1.6 CLAPS (Continued)

as well as conventional eleastic and inelastic buckling analyses of structural components subjected to mechanical loading.

4.1.4.1.7 ASIST

The ASIST program is a General Electric code which can be used to obtain load distribution, deflections, critical frequencies and mode shapes in the "in-plane" or "normal-to-plane" modes for planar structures of any orientation that: (1) are statistically indeterminate; (2) can be represented by straight or curved beams; and (3) are under basically any loading, thermal gradient, or sinusoidal excitation. Deformations and resulting load distributions are compared considering all strain energies (i.e., bending, torsion, shear and direct). ASIST also considers the effects of the deflected shape on loads and provides deflections calculated for the structure. In addition to this beam column (large deflection) capability, the buckling instability of planar structures can also be calculated for the structure. In addition to this beam column (large deflection) capability, the buckling instability of planar structures can also be calculated.

The ASIST program has been used to determine spring constants, stresses, deflections, critical frequencies and associated modes shapes for frames, shafts, rotors, and other jet engine components. It has been used extensively as a design and analysis tool for various components of nuclear fuel assemblies.

4.1.4.2 Fuel Rod Thermal Analysis

The fuel rod thermal analyses models are documented in Section 2 of Reference 2.

4.1.4.3 Reactor Systems Dynamics

The analysis techniques and computer codes used in reactor systems dynamics are described in Section 4 of Reference 1. Su… -tion 4.4.4.6 also provides a complete stability analysis for the reactor coolant system.]

4.1.4.4 Nuclear Analysis

The analysis techniques are described and referenced in Section 3 of Reference 2.]

4.1.4.5 Neutron Fluence Calculations

Neutron vessel fluence calculations were carried out using a one-dimensional, discrete ordinates, Sn transport code with general anisotropic scattering.

This code is a modification of a widely used discrete ordinates code which will solve a wide variety of radiation transport problems. The program will solve both fixed source and multiplication problems. Slab, cylinder, and spherical geometry are allowed with various boundary conditions. The fluence calculations incorporate, as an initial starting point, neutron fission distributions prepared from core physics data as a distributed source. Anisotropic scattering was considered for all regions. The cross sections were prepared with 1/E flux weighted, P sub (L) matrices for anistropic scattering but did not include resonance self-shielding factors. Fast neutron fluxes at locations other than the core mid-plane were calculated using a two-dimensional, discrete ordinate code. The two-dimensional code is an extension of the one-dimensional code.

4.1.4.6 Thermal-Hydraulic Calculations

Description of the thermal-hydraulic models are provided in Section 4 of Reference 2.

4.1.5 References

1. L. A. Carmichael and G. J. Scatena, "Stability and Dyanmic Performance of the General Electric Boiling Water Reactor," January 1977 (NEDO-21506).

2. "General Electric Standard Application for Reactor Fuel," (NEDE-24011-P-A, latest approved revision).

SECTION 4.3

CONTENTS

SECTION 4.3

TABLES

ILLUSTRATIONS

4.3 NUCLEAR DESIGN

See Appendix A, Section A.4.3 of Reference 1.

4.3.1 Design Bases

See Appendix A, Subsection A.4.3.1 of Reference 1.

4.3.2 Description

See Appendix A, Subsection A.4.3.2 of Reference 1.

4.3.2.1 Nuclear Design Description

See Appendix A, Subsection A.4.3.2.1 of Reference 1. The reference core loading pattern for the initial core is to be provided by the applicant as shown in Figure 4.3-1. A summary of the fuel bundles loaded is shown in Table 4.3-1.

4.3.2.2 Power Distribution

See Appendix A, Subsection A.4.3.2.2 of Reference 1.

4.3.2.2.1 Power Distr: :tion Calculations

See Appendix A, Subsection A.4.3.2.2.1 of Reference 1.

A full range of calculated power distributions along with the resultant exposure shapes and the corresponding control rod patterns are shown in Appendix 4A for a typical BWR/6.

4.3.2.2.2 Power Distribution Measurements

See Appendix A, Subsection A.4.3.2.2.2 of Reference 1.

4.3.2.2.3 Power Distribution Accuracy

See Appendix A, Subsection A.4.3.2.2.3 of Reference 1.

4.3.2.2.4 Power Distribution Anomalies

Stringent inspection procedures are utilized to ensure the correct rearrangement of the core following refueling. Although a misplacement of a bundle in the core would be a very improbable event, calculations have been performed in order to determine the effects of such accidents on linear heat generation rate (LHGR) and critical power ratio (CPR). These results are presented in Chapter 15.

The inherent design characteristics of the BWR are well suited to limit gross power tilting. The stabilizing nature of the large moderator void coefficient effectively reduces perturbations in the power distribution. In addition, the in-core instrumentation system, together with the on-line computer, provides the operator with prompt information on power distribution so that he can readily use control rods or other means to limit the undesirable effects of power tilting. Because of these design characteristics, it is not necessary to allocate a specific margin in the peaking factor to account for power tilt. If, for some reason, the power distribution could not be maintained within normal limits using control rods, then the operating power limits would have to be reduced as prescribed in Chapter 16 (Technical Specifications).

4.3.2.3 Reactivity Coefficients

See Appendix A, Subsection A.4.3.2.3 of Reference 1.

4.3.2.4 Control Requirements

See Appendix A, Subsection A.4.3.2.4 of Reference 1.

4.3.2.4.1 Shutdown Reactivity

To assure that the safety design basis for shutdown is satisfied, an additional design margin is adopted: k-effective is calculated to be less than or equal to 0.99 with the control rod highest worth fully withdrawn.

The cold shutdown margin for the reference core loading pattern is given in Table 4.3-2.

4.3.2.4.2 Reactivity Variations

The excess reactivity designed into the core is controlled by the control rod system supplemented by gadolinia-urania fuel rods. The gadolinia-urania concentrations for each fuel type are given in Section 2 of Reference 1.

Control rods are used during the cycle partly to compensate for burnup and partly to flatten the power distribution.

Reactivity balances are not used in describing BWR behavior because of the strong interdependence of the individual constituents of reactivity. Therefore, the design process does not produce components of a reactivity balance at the conditions of interest. Instead, it gives the k_{eff} (Table 4.3-2) representing all effects combined. Further, any listing of components of a reactivity balance is quite ambiguous unless the sequence of the changes is clearly defined.

4.3.2.5 Control Rod Patterns and Reactivity Worths

See Appendix A, Subsection A.4.3.2.5 of Reference 1.

4.3.2.6 Criticality of Reactor During Refueling

See Appendix A, Subsection A.4.3.2.5 of Reference 1.

4.3.2.7 Stability

See Appendix A, Subsection A.4.3.2.6 of Reference 1.

4.3.2.8 Vessel Irradiations

The neutron fluxes at the vessel have been calculated using the one-dimensional discrete ordinates transport code described in Subsection 4.1.4.5. The discrete ordinates code was used in a distributed source mode with cylindrical geometry. The geometry described six regions from the center of the core to a point beyond the vessel. The core region was modeled as a single homogenized cylindrical region. The coolant water region between the fuel channel and the shroud was described containing saturated water at 550°F and 1050 psi. The material compositions for the stainless steel in the shroud and the carbon steel in the vessel contain the mixtures by weight as specified in the ASME material specifications for ASME SA 240, 304L, and ASME SA 533 grade B. In the region between the shroud and the vessel, the presence of the jet pumps was ignored. A simple diagram showing the regions, dimensions, and weight fractions are shown in Figure 4.3-2.

The distributed source used for this analysis was obtained from the gross radial power description. The distributed source at any point in the core is the product of the power from the power description and the neutron yield from fission. By using the neutron energy spectrum, the distributed source is obtained for position and energy. The integral over position and energy is normalized to the total number of neutrons in the core region. The core region is defined as a 1 centimeter thick disc with no transverse leakage. The power in this core region is set equal to the maximum power in the axial direction.

4.3.2.8 Vessel Irradiations (Continued)

The neutron fluence is determined from the calculated flux by assuming that the plant is operated 90 percent of the time at 90 percent power level for 40 years or equivalent to 1×10^9 full power seconds. The calculated fluxes and fluence are shown in Table 4.3-3. The calculated neutron flux leaving the cylindrical core is shown in Table 4.3-4.

4.3.3 Analytical Methods

See Appendix A, Subsection A.4.3.3 of Reference 1.

4.3.4 Changes

See Appendix A, Subsection A.4.3.4 of Reference 1.

4.3.5 References

1. "General Electric Standard Application for Reactor Fuel," (NEDE-24011-P-A, latest approved revision).

Table 4.3-1

REFERENCE CORE LOADING PATTERN

Fuel Designation*	Number Loaded
(Provided by Applicant)	(Provided by Applicant)

*Reference 1 nomenclature

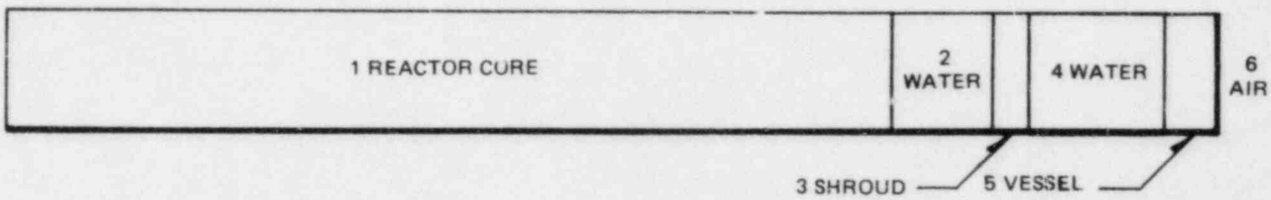
Table 4.3-4

CALCULATED NEUTRON FLUX AT CORE EQUIVALENT BOUNDARY

Group	Lower Energy Bound (eV)	Flux (n/cm^2-sec)
1	10.0×10^6	3.6×10^{10}
2	6.065×10^6	5.3×10^{11}
3	3.679×10^6	2.0×10^{12}
4	2.231×10^6	3.9×10^{12}
5	1.353×10^5	4.6×10^{12}
6	8.208×10^5	4.1×10^{12}
7	4.979×10^5	4.0×10^{12}
8	3.020×10^5	2.8×10^{12}
9	1.832×10^5	2.4×10^{12}
10	6.738×10^4	3.4×10^{12}
11	2.479×10^4	2.3×10^{12}
12	9.119×10^3	2.3×10^{12}
13	3.355×10^3	2.1×10^{12}
14	1.234×10^3	2.1×10^{12}
15	4.540×10^2	2.0×10^{12}
16	1.670×10^2	2.1×10^{12}
17	6.144×10^1	1.9×10^{12}
18	2.260×10^1	1.9×10^{12}
19	1.371×10^1	9.2×10^{12}
20	8.315	9.2×10^{12}
21	5.043	8.4×10^{12}
22	3.059	8.7×10^{11}
23	1.255	8.6×10^{11}
24	1.125	8.5×10^{11}
25	0.616	9.1×10^{11}
26	0.000	3.2×10^{13}

(Provided by Applicant)

Figure 4.3-1. Reference Core Loading Pattern



MATERIAL		RADIUS (inches)	MATERIAL	VOLUME AVERAGE DENSITY
NO.	NAME			
1	REACTOR CORE	92.58	WATER UO_2 304L STAINLESS STEEL ZIRCONIUM	0.318 g/cm^3 2.334 g/cm^3 0.056 g/cm^3 0.978 g/cm^3
2	WATER	99.9	WATER	0.74 g/cm^3
3	SHROUD	101.9	304L STAINLESS STEEL	FROM ASME SA 240
4	WATER	119.0	WATER	0.74 g/cm^3
5	VESSEL	125.0	CARBON STEEL	FROM ASME SA 533
6	AIR		AIR	1.3×10^{-3} g/cc

Figure 4.3-2. Model for One-Dimensional Transport Analysis of Vessel Fluence

Table 4.4-1

THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS
OF THE REACTOR CORE

	218-624	238-748	251-800
General Operating Conditions			
Reference rated thermal output (MWt)	2,894	3,579	3,833
Design power level for engineered safety features (MWt)	3,016	3,730	3,99[illegible]
Rated steam flow rate, at 420°F final feedwater temperature (millions lb/hr)	12.453	15.400	16.49
Core coolant flow rate (millions lb/hr)	84.5	104.0	112.5
Feedwater flow rate (millions lb/hr)	12.428	15.367	16.46
System pressure, nominal in steam dome (psia)	1,040	1,040	1,040
System pressure, nominal core design (psia)	1,055	1,055	1,055
Coolant saturation temperature at core design pressure (°F)	551	551	551
Average power density (kW/liter)	52.4	54.1	54.1
Maximum LHGR (kW/ft)	13.4	13.4	13.4
Average LHGR (kW/ft)	5.7	5.9	5.9
Core total heat transfer area (ft^2)	61,151	73,303	78,398
Maximum heat flux ($Btu/hr\text{-}ft^2$)	361,600	361,600	361,600
Average heat flux ($Btu/hr\text{-}ft^2$)	154,600	159,500	159,800
Design operating MCPR	See Table 15.0-2		

4.1

Table 4.4-1

THERMAL AND HYDRAULIC DESIGN CHARACTERISTICS
OF THE REACTOR CORE (Continued)

	218-624	238-748	251-800
General Operating Conditions (Continued)			
Core inlet enthalpy at 420°F FFWT (Btu/lb)	527.8	527.7	527.9
Core inlet temperature, at 420°F FFWT, (°F)	533	533	533
Core maximum exit voids within assemblies (%)	76.0	79.0	76.0
Core average void fraction, active coolant	0.411	0.414	0.412
Maximum fuel temperature (°F)	3,435	3,435	3,435
Active coolant flow area per assembly (in.^{2})	15.164	15.164	15.164
Core average inlet velocity (ft/sec)	6.82	6.98	7.07
Maximum inlet velocity (ft/sec)	7.90	8.54	8.57
Total core pressure drop (psi)	25.26	26.4	26.74
Core support plate pressure drop (psi)	20.84	22.0	22.32
Average orifice pressure drop			
Central region (psi)	5.41	5.71	5.78
Peripheral region (psi)	17.95	18.68	19.16
Maximum channel pressure loading (psi)	14.52	15.40	15.59
Average-power assembly channel pressure loading (bottom) (psi)	13.28	14.1	14.22
Shroud support ring and lower shroud pressure loading (psi)	24.84	25.7	25.12
Upper shroud pressure loading (psi)	4.0	3.7	2.8

4.1

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Table 4.4-6

LENGTHS OF SAFETY INJECTION LINES

Loop	Line	Nominal Diameter (in)	Pipe Schedule	Length (ft)

(Provided by Applicant)

APPENDIX 4A

CONTENTS

APPENDIX 4A

ILLUSTRATIONS (Continued)

4A.2 POWER DISTRIBUTION STRATEGY

A basic operating principle used to minimize power peaking throughout an operating cycle has been developed and is applied to boiling water reactors. The principle, the Haling principle, is described in Reference 1. The main concept is that "for any given set of end-of-cycle conditions, the power peaking factor is maintained at the minimum value when the power shape does not change during the operating cycle".

4A.3 RESULTS OF CORE SIMULATION STUDIES

The following table itemizes the exposure step and its related figure numbers:

Incremental Exposure (GWd/st)	Sequence*	Figure Numbers
6.69	All-rods-out - Haling EOC	4A-1a through 4A-1d
0.2	A-2	4A-2a through 4A-2e
1.0	B-2	4A-3a through 4A-3e
2.0	A-1	4A-4a through 4A-4e
3.0	B-1	4A-5a through 4A-5e
4.0	A-2	4A-6a through 4A-6e
5.0	B-2	4A-7a through 4A-7e
6.0	A-1	4A-8a through 4A-8e
6.6	All rods out	4A-9a through 4A-9e

The detailed data presented demonstrates that this design can be operated throughout this cycle with adequate margins to allow for operating flexibility. The variation of the maximum linear heat generation rate (MLHGR) with cycle exposure is presented in Figure 4A-10. Significant margin exists relative to the MLHGR operating limit. Maximum average planar linear heat generation rates (MAPLHGR) are not calculated for this design since calculations show the peak clad temperature (PCT) to be less than the 2200°F limit when the maximum single rod is at the 13.4 kW/ft limit. Adherence to the MLHGR limit will always assure meeting the MAPLHGR limit. The variation of the minimum critical power ratio (MCPR) with cycle exposure is shown in Figure 4A-11. Similarly, a large margin is indicated with respect to the expected MCPR operating limit.

I	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
J															
1										0.3003	0.3751	0.4104	0.4224	0.4272	0.4216
2								0.3264	0.4470	0.6476	0.7892	0.8983	0.8501	0.9179	0.8340
3							0.3973	0.7105	0.8897	0.8405	1.0449	0.9271	1.0938	0.9306	0.9846
4						0.4135	0.7566	0.9627	0.9058	1.1126	0.9967	1.1841	1.0292	1.1840	0.9919
5					0.4228	0.7741	0.9921	0.9291	1.0382	1.0492	1.1179	1.0873	1.1582	1.0837	1.1220
6				0.4169	0.7790	1.0108	0.9478	1.1650	1.0415	1.2384	1.0224	1.2853	1.1098	1.2855	1.0328
7			0.4015	0.7645	1.0059	1.0239	1.0607	1.0728	1.1411	1.1122	1.1962	1.1506	1.2385	1.1460	1.1783
8		0.3299	0.7175	0.9717	0.9383	1.1657	0.9693	1.2345	1.0273	1.2973	1.0679	1.3422	1.1495	1.3314	1.0479
9		0.4507	0.8980	0.9138	1.0451	1.0396	1.1325	1.0255	1.1956	1.1518	1.2314	1.1653	1.2452	1.1481	1.1739
10	0.3038	0.6522	0.8489	1.1252	1.0482	1.2477	1.1135	1.2993	1.1351	1.3368	1.0769	1.3257	1.0722	1.3017	1.0298
11	0.3789	0.7970	1.0573	1.0108	1.1363	1.0354	1.2034	1.0739	1.2427	1.0854	1.2336	1.0652	1.2242	1.1085	1.1411
12	0.4152	0.9065	0.9373	1.1993	1.0999	1.3049	1.1610	1.3530	1.1803	1.3502	1.1549	1.3026	1.0385	1.2438	0.9801
13	0.4254	0.8553	1.1027	1.0377	1.1670	1.1156	1.2403	1.1534	1.2547	1.0836	1.2228	1.0462	1.1834	0.9901	1.0766
14	0.4291	0.9217	0.9336	1.1891	1.0875	1.2876	1.1448	1.3304	1.1540	1.3135	1.1235	1.2648	1.0663	1.1812	0.9185
15	0.4221	0.8348	0.9860	0.9923	1.1190	1.0310	1.1779	1.0487	1.1326	1.0378	1.1473	0.9920	1.0890	0.9220	0.9463

Figure 4A-9d. Integrated Power per Bundle at 6.6 GWd/st Cycle Exposure

I	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
J															
1										25451.2	24479.5	24674.7	24815.8	24845.5	24938.7
2								25421.8	25008.9	3986.5	13835.1	5564.2	14365.2	5949.9	14615.4
3							24730.8	13488.2	5562.5	21248.2	6521.4	21941.6	7072.6	22318.8	15111.2
4						25454.8	13930.5	6198.4	20959.7	7230.6	21849.8	7636.5	22205.5	7879.8	22964.1
5					24960.7	14147.2	6456.7	22129.9	16057.2	20076.8	15368.1	21107.3	14923.6	21578.1	15781.5
6				25454.2	14110.3	6583.4	21864.7	7771.8	22141.2	8220.4	28390.7	8231.0	23279.3	8138.1	26609.4
7			24720.0	13907.1	6488.4	15959.0	16142.2	20022.9	15651.3	21540.0	14774.7	21506.6	14110.2	21493.9	15664.3
8		25320.8	13411.0	6124.1	21760.8	7660.9	28696.2	8005.6	28590.3	8100.6	28381.7	7802.9	23113.6	7646.6	27504.0
9		24979.5	5453.4	20828.6	15889.5	22298.0	15536.4	28565.3	14787.8	21388.5	14386.4	21633.2	14024.6	21715.0	16261.0
10	25284.8	3896.0	21117.1	7026.1	20653.9	7961.1	21486.9	8007.8	22964.5	7551.7	28395.6	7773.1	28031.4	7956.0	27843.9
11	24417.0	13724.6	6360.5	21505.3	14817.1	27894.4	14736.8	28240.4	14039.5	28136.9	13873.8	28543.2	12763.1	22294.5	16229.8
12	24443.1	5465.6	21756.3	7407.2	20950.7	7695.7	21294.4	7692.4	21323.0	7469.9	21729.8	7865.1	28432.7	8212.6	28683.6
13	24781.0	14323.9	6972.3	22076.4	14833.1	23267.4	14308.8	23218.2	13994.0	28016.8	14038.6	28243.7	12814.2	28443.5	16382.4
14	24827.9	5904.2	22304.3	7835.0	21517.7	8182.1	21705.3	7910.7	21717.8	7847.4	21874.4	7979.9	22272.3	8074.4	28690.5
15	24946.7	14596.9	15061.0	22982.1	16055.7	26829.3	15714.7	27506.3	15814.8	27365.3	16151.4	28011.3	16107.2	28452.3	21915.2

Figure 4A-9e. Average Bundle Exposure at 6.6 GWd/st Cycle Exposure

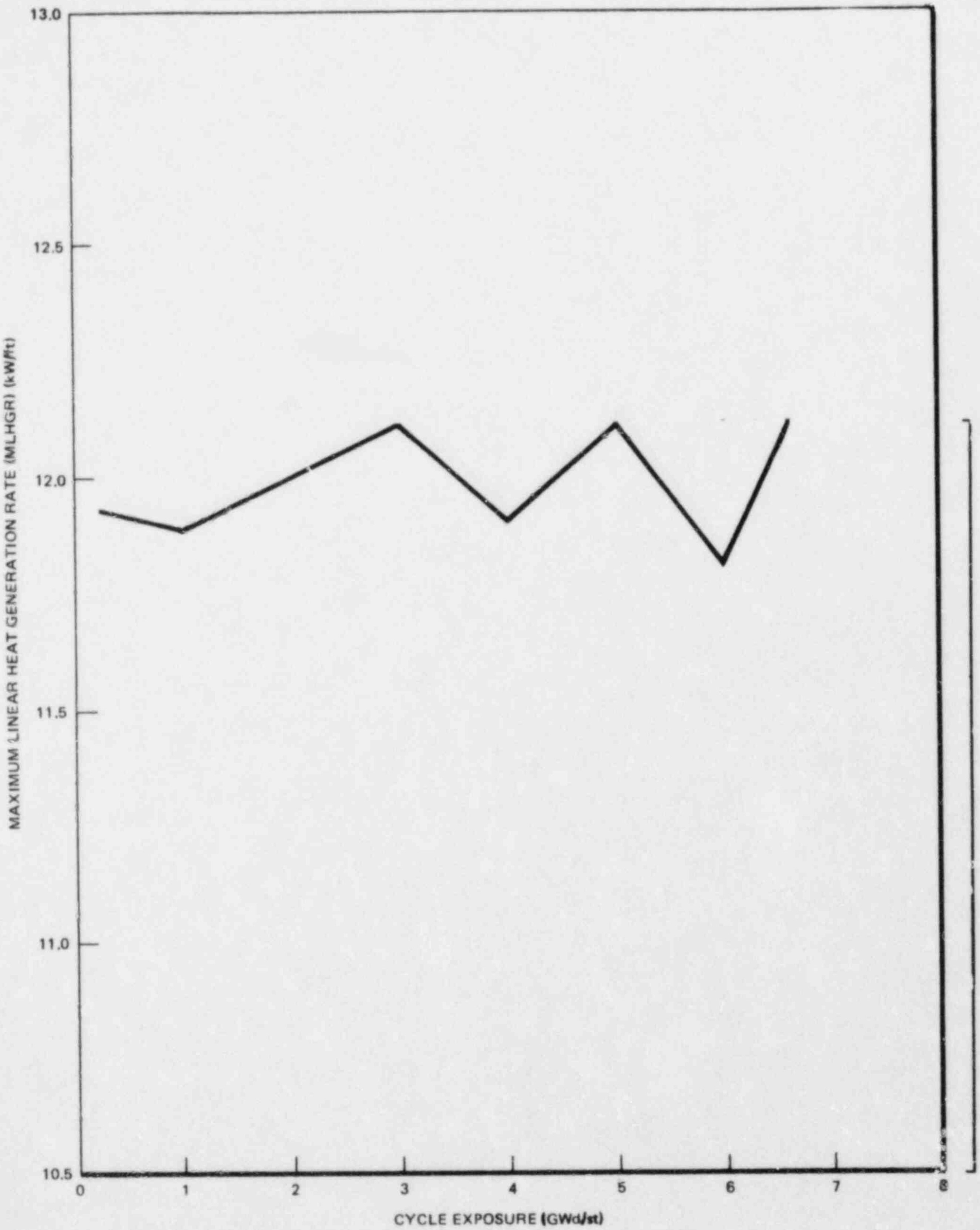

Figure 4A-10. Maximum Linear Heat Generation Rate as a Function of Cycle Exposure

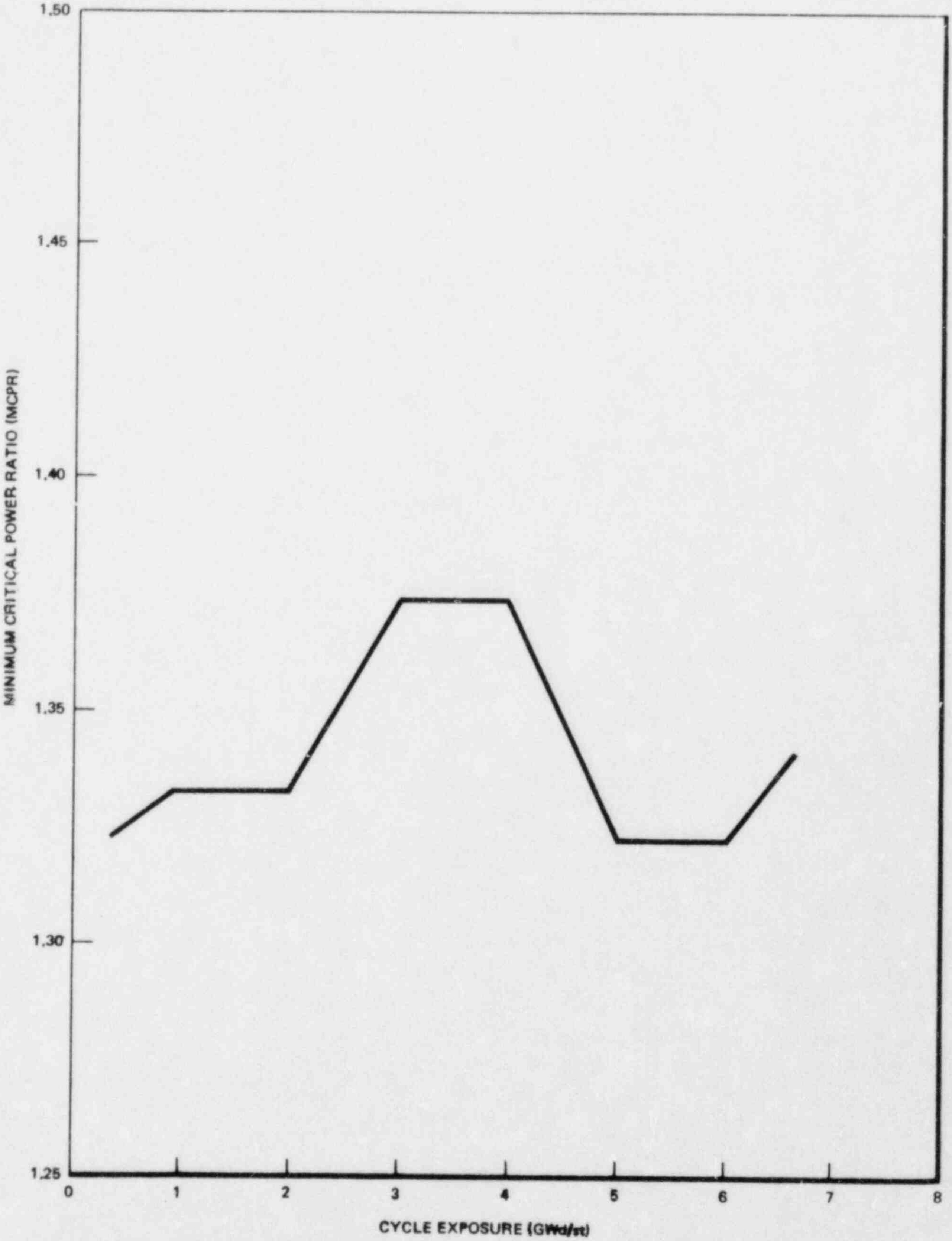

Figure 4A-11. Minimum Critical Power Ratio as a Function of Cycle Exposure

6.3.2.8 Manual Actions (Continued)

as indicating the operation of the ECCS. ECCS flow indication is the primary parameter available to assess proper operation of the system. Other indications, such as position of valves, status of circuit breakers, and essential power bus voltage, are available to assist him in determining system operating status. The electrical and instrumentation complement to the ECCS is discussed in detail in Section 7.3. Other available instrumentation is listed in the P&IDs for the individual systems. Much of the monitoring instrumentation available to the operator is discussed in more detail in Chapter 5 and Section 6.2.

6.3.3 ECCS Performance Evaluation

The performance of the ECCS is determined through application of the 10CFR50 Appendix K evaluation models and then showing conformance to the acceptance criteria of 10CFR50.46. Analytical models are documented in Subsection S.2.5.2 of Reference 4.

The ECCS performance is evaluated for the entire spectrum of break sizes for postulated LOCAs. MAPLHGR results are for a fuel enrichment of approximately 3 wt% U-235.

The accidents, as listed in Chapter 15, for which ECCS operation is required are:

Subsection	Title
15.2.8	Feedwater Piping Break
15.6.4	Spectrum of BWR Steam System Piping Failures Outside of Containment
15.6.5	Loss-of-Coolant Accidents

Chapter 15 provides the radiological consequences of the above listed events.

6.3.3.1 ECCS Bases for Technical Specifications

The maximum average planar linear heat generation rates (MAPLHGR) calculated in this performance analysis provide the basis for Technical Specifications designed to ensure conformance with the acceptance criteria of 10CFR50.46. Minimum ECCS functional requirements are specified in Subsections 6.3.3.4 and 6.3.3.5, and testing requirements are discussed in Subsection 6.3.4. Limits on minimum suppression pool water level are discussed in Section 6.2.

6.3.3.2 Acceptance Criteria for ECCS Performance

The applicable acceptance criteria, extracted from 10CFR50.46 are listed, and, for each criterion, applicable parts of Subsection 6.3.3 (where conformance is demonstrated) are indicated. A detailed description of the methods used to show compliance are shown in Subsection S.2.5.2 of Reference 4.

Criterion 1: Peak Cladding Temperature

"The calculated maximum fuel element cladding temperature shall not exceed 2200°F." Conformance to Criterion 1 is shown in Subsections 6.3.3.7.3 (Break Spectrum), 6.3.3.7.4 (Design Basis Accident), 6.3.3.7.5 (Transition Break), 6.3.3.7.6 (Small Break), and specifically in Table 6.3-4 (MAPLHGR, maximum local oxidation, and peak cladding temperature versus exposure).

Criterion 2: Maximum Cladding Oxidation

"The calculated total local oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation." Conformation to Criterion 2 is shown in Figure 6.3-8 (break spectrum plot), Table 6.3-4 (local oxidation versus exposure) and Table 6.3-5 (break spectrum summary).

6.3.3.2 Acceptance Criteria for ECCS Performance (Continued)

Criterion 3: Maximum Hydrogen Generation

"The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinder surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react." Conformance to Criterion 3 is shown in Table 6.3-5.

Criterion 4: Coolable Geometry

"Calculated changes in core geometry shall be such that the core remains amenable to cooling." As described in Reference 2, Section III.A, conformance to Criterion 4 is demonstrated by conformance to Criterion 1 and 2.

Criterion 5: Long-Term Cooling

"After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core." Conformance to Criterion 5 is demonstrated generically for General Electric BWRs in Reference 2, Section III.A. Briefly summarized, the core remains covered to at least the jet pump suction elevation and the uncovered region is cooled by spray cooling and/or by steam generated in the covered part of the core.

6.3.3.3 Single-Failure Considerations

The functional consequences of potential operator errors and single failures (including those which might cause any manually

6.3.3.3 Single-Failure Considerations (Continued)

controlled electrically operated valve in the ECCS to move to a position which could adversely affect the ECCS) and the potential for submergence of valve motors in the ECCS are discussed in Subsection 6.3.2. There it was shown that all potential single failures are no more severe than one of the single failures identified in Table 6.3-3.

It is therefore only necessary to consider each of these single failures in the ECCS performance analyses. For large breaks, failure of one of the diesel generators is, in general, the most severe failure. For small breaks, the HPCS is the most severe failure.

A single failure in the ADS (one ADS valve) has no effect in large breaks. Therefore, as a matter of calculational convenience, it is assumed in all calculations that one ADS valve fails to operate in addition to the identified single failure. This assumption reduces the number of calculations required in the performance analysis and bounds the effects of one ADS valve failure and HPCS failure by themselves. The only effect of the assumed ADS valve failure by the calculations is a small increase (on the order of 100°F) in the calculated temperatures following small breaks.

6.3.3.4 System Performance During the Accident

In general, the system response to an accident can be described as:

(1) receiving an initiation signal;

(2) a small lag time (to open all valves and have the pumps up to rated speed); and

(3) finally, the ECCS flow entering the vessel.

6.3.3.4 System Performance During the Accident (Continued)

Key ECCS actuation setpoints and time delays for all the ECCS systems are provided in Table 6.3-1. The minimization of the delay from the receipt of signal until the ECCS pumps have reached rated speed is limited by the physical constraints on accelerating the diesel-generators and pumps. The delay time due to valve motion in the case of the high pressure system provides a suitably conservative allowance for valves available for this application. In the case of the low pressure system, the time delay for valve motion is such that the pumps are at rated speed prior to the time the vessel pressure reaches the pump shutoff pressure.

The flow delivery rates analyzed in Subsection 6.3.3 can be determined from the head-flow curves in Figures 6.3-3, 6.3-4 and 6.3-5 and the pressure versus time plots discussed in Subsection 6.3.3.7. Simplified piping and instrumentation and process diagrams for the ECCS are referenced in Subsection 6.3.2. The operational sequence of ECCS for the DBA is shown in Table 6.3-2.

Operator action is not required, except as a monitoring function, during the short-term cooling period following the LOCA. During the long-term cooling period, the operator will take action as specified in Subsection 6.2.2.2 to place the containment cooling system into operation.

6.3.3.5 Use of Dual Function Components for ECCS

See Appendix A, Subsection A.6.3.3.5 of Reference 4.

6.3.3.6 Limits on ECCS System Parameters

See Appendix A, Subsection A.6.3.3.6 of Reference 4.

6.3.3.7 ECCS Analyses for LOCA

6.3.3.7.1 LOCA Analysis Procedures and Input Variables

See Appendix A, Subsection A.6.3.3.7.1 of Reference 4. The significant input variables used by the LOCA codes are listed in Table 6.3-1 and Figure 6.3-9.

6.3.3.7.5 Transition Recirculation Line Break Calculations (Continued)

(11) fuel rod convective heat transfer coefficients (small break methods) as a function of time; and

(12) peaking cladding temperature (small break methods) as a function of time.

6.3.3.7.6 Small Recirculation Line Break Calculations

Important variables from the analysis of the small break yielding the highest cladding temperature are shown in Figures 6.3-48 through 6.3-51. These variables are:

(1) water level as a function of time;

(2) pressure as a function of time;

(3) convective heat transfer coefficients as a function of time; and

(4) peak cladding temperature as a function of time.

The same variables resulting from the analysis of a less limiting small break are shown in Figures 6.4-52 through 6.3-55.

6.3.3.7.7 Calculations for Other Break Locations

Reactor water level and vessel pressure and peak cladding temperature and fuel rod convective heat transfer coefficients are shown in Figures 6.3-56 through 6.3-59 for the core spray line break, Figures 6.3-60 through 6.3-63 for the feedwater line break, and in Figures 6.3-64 and 6.3-65 for the main steamline break inside the containment.

6.3.3.7.7 Calculations for Other Break Locations (Continued)

An analysis was also done for the main steamline break outside the containment. Reactor water level and vessel pressure and peak cladding temperature and fuel rod convective heat transfer coefficients are shown in Figures 6.3-68 through 6.3-71.

6.3.3.7.8 Improved Decay Heat Correlation

Section I.A.4 of 10CFR50, Appendix K, requires use of the 1971 ANS Standards Subcommittee proposed decay heat standard for ECCS licensing evaluations. The current method for applying the 1971 standards in BWR LOCA calculations is outlined in GE's approved ECCS evaluation model (Reference 2). In 1979, the American National Standards Institute approved and the ANS published a much improved decay heat standard (Reference 3). A detailed technical basis for an improved GE BWR decay heat correlation based on the 1979 standard is outlined in Appendix 6A. Use of the improved correlation in the currently approved GE LOCA models will provide increased ECCS criteria margins.

Application of the correlation described in Appendix 6A is optional. To use it in place of the current method, a utility must provide the NRC with a request for exemption from Section I.A.4 of 10CFR50, Appendix K. The utility must reference Appendix 6A as the technical justification for the exemption.

6.3.3.8 LOCA Analysis Conclusions

Having shown compliance with the applicable acceptance criteria of Section 6.3.3.2, it is concluded that the ECCS will perform its function in an acceptable manner and meet all of the 10CFR50.46 acceptance criteria, given operation at or below the MAPLHGRs in Table 6.3-4.

6.3.4.2.4 LPCI Testing

Each LPCI loop can be tested during reactor operation. The test conditions are tabulated in Figures 6.3-4a, b and c. During plant operation, this test does not inject cold water into the reactor because the injection line check valve is held closed by vessel pressure, which is higher than the pump pressure. The injection line portion is tested with reactor water when the reactor is shut down and when a closed system loop is created. This prevents unnecessary thermal stresses.

To test an LPCI pump at rated flow, the test line valve to the suppression pool is opened, the pump suction valve from the suppression pool is opened (this valve is normally open) and the pumps are started using the remote/manual switches in the control room. Correct operation is determined by observing the instruments in the control room.

If an initiation signal occurs during the test, the LPCI System returns to the operating mode. The valves in the test bypass lines are closed automatically to assure that the LPCI pump discharge is correctly routed to the vessel.

6.3.5 Instrumentation Requirements

Design details including redundancy and logic of the ECCS instrumentation are discussed in Section 7.3.

All instrumentation required for automatic and manual initiation of the HPCS, LPCS, LPCI and ADS is discussed in Subsection 7.3.1 and is designed to meet the requirements of IEEE-279 and other applicable regulatory requirements. The HPCS, LPCS, LPCI and ADS can be manually initiated from the control room.

6.3.5 Instrumentation Requirements (Continued)

The HPCS, LPCS and LPCI are automatically initiated on low reactor water level or high drywell pressure. (See Table 6.3-8 for specific initiation levels for each system) The ADS is automatically actuated by sensed variables for reactor vessel low water level and drywell high pressure plus indication that at least one LPCI or LPCS pump is operating. The HPCS, LPCS and LPCI automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initiation signal. The LPCS and LPCI system injection into the RPV begin when reactor pressure decreases to system discharge shutoff pressure.

HPCS injection begins as soon as the HPCS pump is up to speed and the injection valve is open, since the HPCS is capable of injecting water into the RPV over a pressure range from 1177 psid* to 200 psid[3].

6.3.6 References

1. H.M. Hirsch, "Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems", January 1973 (NEGO-10739).

2. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix", November 1975 (NEDE-20566P).

3. "Decay Heat Power in Light Water Reactors", ANSI/ANS 5.1-1979, Approved by American National Standards Institute, August 29, 1979.

4. "General Electric Standard Application for Reactor Fuel-United States Supplement", NEDE-20411-P-A-US (latest approved revision).

[3]psid - differential pressure between RPV and pump suction source.

SECTION 15.1

TABLES

ILLUSTRATIONS

15.1.4.2.2 Systems Operation

This event assumes normal functioning of normal plant instrumentation and controls, specifically the operation of the pressure regulator and level control systems.

15.1.4.3 Core and System Performance

The opening of a S/R valve allows steam to be discharged into the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a mild depressurization transient.

The pressure regulator senses the nuclear system pressure decrease and within a few seconds closes the turbine control valve far enough to stabilize reactor vessel pressure at a slightly lower value and reactor power settles at nearly the initial power level. Thermal margins decrease only slightly through the transient, and no fuel damage results from the transient. MCPR is essentially unchanged and, therefore, the safety limit margin is unaffected and this event does not have to be reanalyzed for specific core configurations.

15.1.4.4 Barrier Performance

As discussed above, the transient resulting from a stuck open relief valve is a mild depressurization which is within the range of normal load following and therefore has no significant effect on RCPB and containment design pressure limits.

15.1.4.5 Radiological Consequences

While the consequences of this event does not result in fuel failure, it does result in the discharge of normal coolant activity to the suppression pool via SRV operation. Since this activity is

15.1.4.5 Radiological Consequences (Continued)

contained in the primary containment, there will be no exposures to operating personnel. Since this event does not result in an uncontrolled release to the environment, the plant operator can choose to leave the activity bottled up in the containment or discharge it to the environment under controlled release conditions. If purging of the containment is chosen, the release will be in accordance with the established technical specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.1.5 Spectrum of Steam System Piping Failures Inside and Outside of Containment in a PWR

This event is not applicable to BWR plants.

15.1.6 Inadvertent RHR Shutdown Cooling Operation

15.1.6.1 Identification of Causes and Frequency Classification

15.1.6.1.1 Identification of Causes

At design power conditions, no conceivable malfunction in the shutdown cooling system could cause temperature reduction.

In startup or cooldown operation, if the reactor were critical or near critical, a very slow increase in reactor power could result. A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for the RHR heat exchangers. The resulting temperature decrease would cause a slow insertion of positive reactivity into the core. If the operator did not act to control the power level, a high neutron flux reactor scram would terminate the transient without violating fuel thermal limits and without any measurable increase in nuclear system pressure.

15.1.6.1.2 Frequency Classification

Although no single failure could cause this event, it is conservatively categorized as an event of moderate frequency.

15.1.6.2 Sequence of Events and Systems Operation

15.1.6.2.1 Sequence of Events

A shutdown cooling malfunction leading to a moderator temperature decrease could result from misoperation of the cooling water controls for RHR heat exchangers. The resulting temperature decrease causes a slow insertion of positive reactivity into the core. Scram will occur before any thermal limits are reached if the operator does not take action. The sequence of events for this event is shown in Table 15.1-6.

15.1.6.2.2 System Operation

A shutdown cooling malfunction causing a moderator temperature decrease must be considered in all operating states. However, this event is not considered while at power operation since the nuclear system pressure is too high to permit operation of the shutdown cooling (RHRs).

No unique safety actions are required to avoid unacceptable safety results for transients as a result of a reactor coolant temperature decrease induced by misoperation of the shutdown cooling heat exchangers. In startup or cooldown operation, where the reactor is at or near critical, the slow power increase resulting from the cooler moderator temperature would be controlled by the operator in the same manner normally used to control power in the source or intermediate power ranges.

15.1.6.3 Core and System Performance

The increased subcooling caused by misoperation of the RHR shutdown cooling mode could result in a slow power increase due to the reactivity insertion. This power rise would be terminated by a flux scram before fuel thermal limits are approached. Therefore, only qualitative description is provided here and this event does not have to be analyzed for specific core configurations.

15.1.6.4 Barrier Performance

As noted above, the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.6.5 Radiological Consequences

Since this event does not result in any fuel failures, no analysis of radiological consequences is required for this event.

Table 15.1-1

SEQUENCE OF EVENTS FOR FIGURE 15.1-1

Time (sec)	Event
0	Initiate a 100°F temperature reduction in the feedwater system.
5	Initial effect of unheated feedwater starts to raise core power level but AFC system automatically reduces core flow to maintain initial steam flow.
100	Reactor variables settle into new steady state.

Table 15.1-2

SEQUENCE OF EVENTS FOR FIGURE 15.1-2

Time (sec)	Event
0	Initiate a 100°F temperature reduction into the feedwater system.
5	Initial effect of unheated feedwater starts to raise core power level and steam flow.
7	Turbine control valves start to open to regulate pressure.
36	APRM initiates reactor scram on high thermal power.
44.0	Narrow Range (NR) sensed water level reaches Level 3 (L3) setpoint. Recirculation pumps tripped to low frequency speed.
>50 (est)	Recirculation Pump Trip initiated due to Level 2 Trip. (not included in simulation).
>50 (est)	Wide Range (WR) sensed water level reaches Level 2 (L2) setpoint.
>80 (est)	HPCS/RCIC flow enters vessel (not simulated).
>90 (est)	Reactor variables settle into limit cycle.

Table 15.1-3

SEQUENCE OF EVENTS FOR FIGURE 15.1-3

Time (sec)	Event
0	Initiate simulated failure of 130% upper limit at system design pressure of 1065 psig on feedwater flow.
11.8	L8 vessel level setpoint initiates reactor scram and trips main turbine and feedwater pumps.
11.9	Recirculation pump trip (RPT) actuated by stop valve position switches.
11.9	Main turbine bypass valves opened due to turbine trip.
13.2	Safety/relief valves open due to high pressure.
18.2	Safety/relief valves close.
>20 (est)	Water level dropped to low water level setpoing (Level 2).
>50 (est)	RCIC and HPCS flow into vessel (not simulated).

Table 15.1-4

SEQUENCE OF EVENTS FOR FIGURE 15.1-4

Time (sec)	Event
0	Simulate steam flow demand to 130%.
2.1	Turbine control valves wide open.
2.28	Vessel water level (L8) trip initiates reactor scram and main turbine and feedwater turbine trips.
2.28	Turbine trip initiates bypass operation to full flow.
2.29	Main turbine stop valves reach 90% open position and initiates recirculation pump trip (RPT).
2.38	Turbine stop valves closed. Turbine bypass valves opening to full flow.
2.4	Recirculation pump motor circuit breakers open causing decrease in case flow to natural circulation.
5.2	Group 1 S/R valves open again to relieve decay heat.
10.2	Group 1 S/R valves close again.
25	Vessel water level reaches L2 setpoint.
28	Low turbine inlet pressure trip initiates main steamline isolation.
33	Main steam isolation valves closed. Bypass valves remain open, exhausting steam in steamlines downstream of isolation valves.
55 (est)	HPCS and RCIC flow enters vessel (not simulated).

Table 15.1-5

SEQUENCE OF EVENTS FOR INADVERTENT SAFETY/RELIEF VALVE OPENING

Time (sec)	Event
0	Initiate opening of 1 S/R valve.
0.5 (est.)	Relief flow reaches full flow.
15 (est.)	System establishes new steady-state operation.

Table 15.1-6

SEQUENCE OF EVENTS FOR INADVERTENT RHR SHUTDOWN COOLING OPERATION

Approximate Elapsed Time	Event
0	Reactor at states B or D (of Appendix 15A) when RHR shutdown cooling inadvertently activated.
0-10 min	Slow rise in reactor power.
+10 min	Operator may take action to limit power rise. Flux scram will occur if no action is taken.

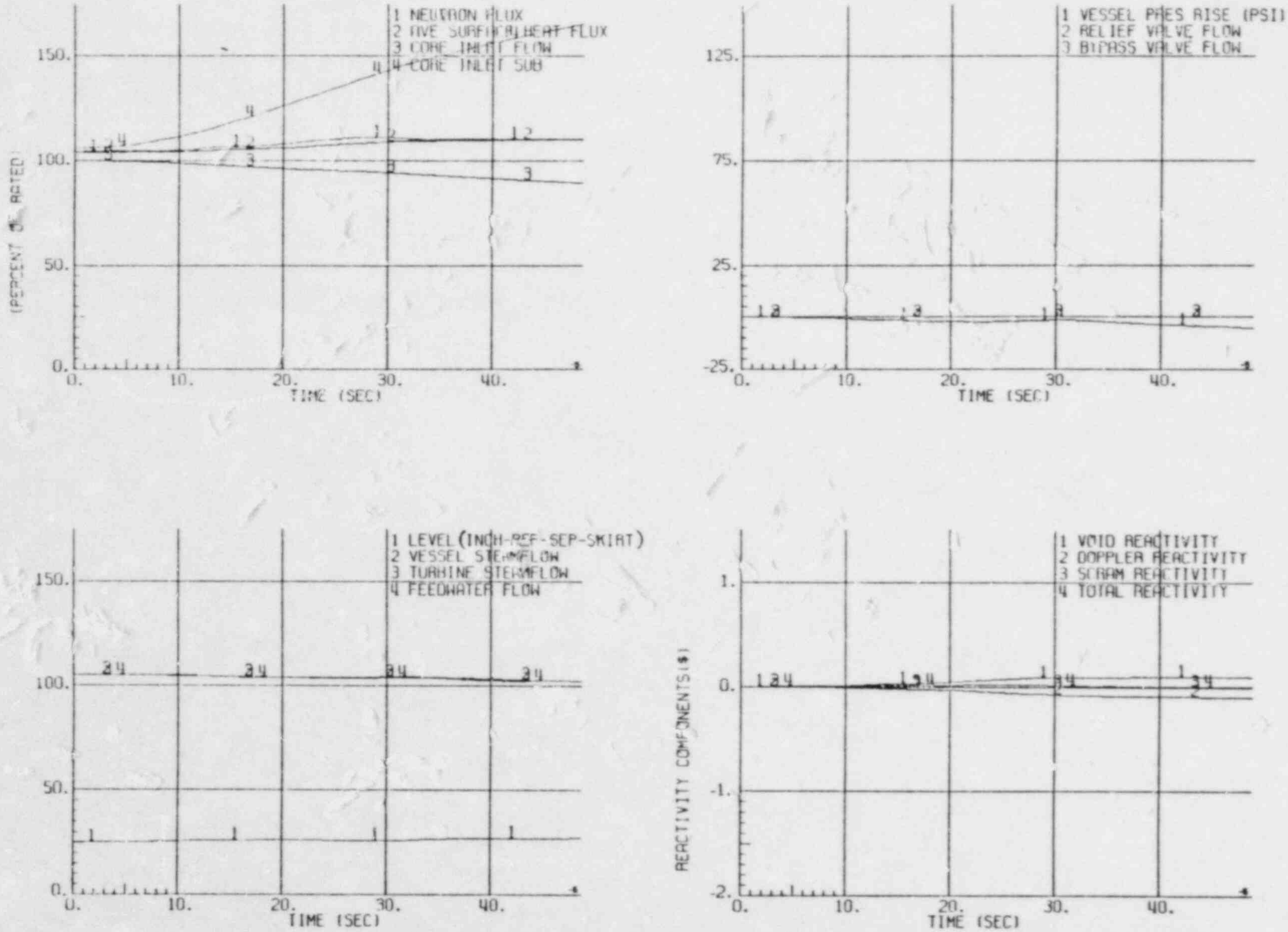

Figure 15.1-1. Loss of 100°F Feedwater Heating
(Automatic Flow Control Mode)

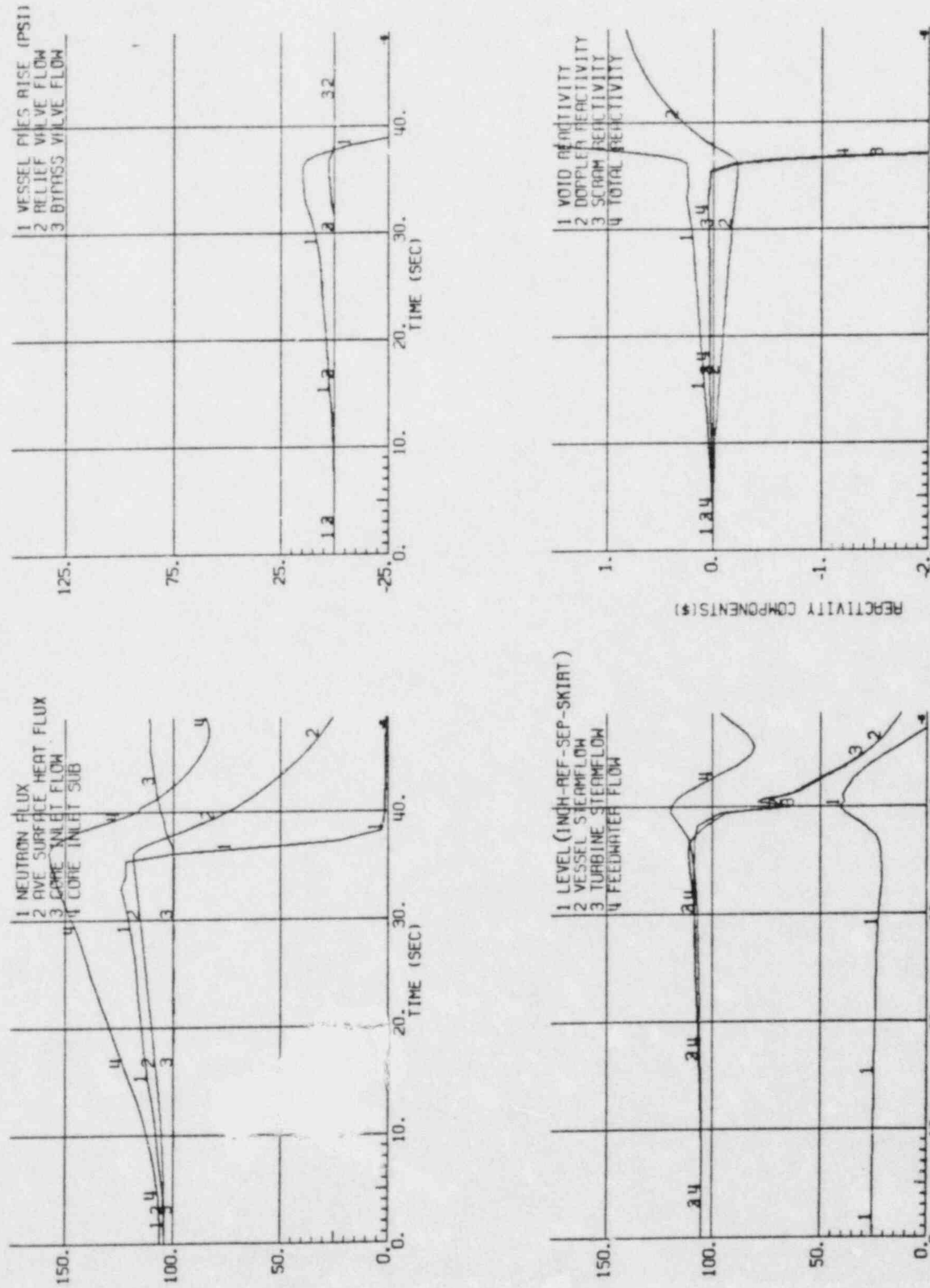

Figure 15.1-2. Loss of 100°F Feedwater Heating (Manual Flow Control Mode)

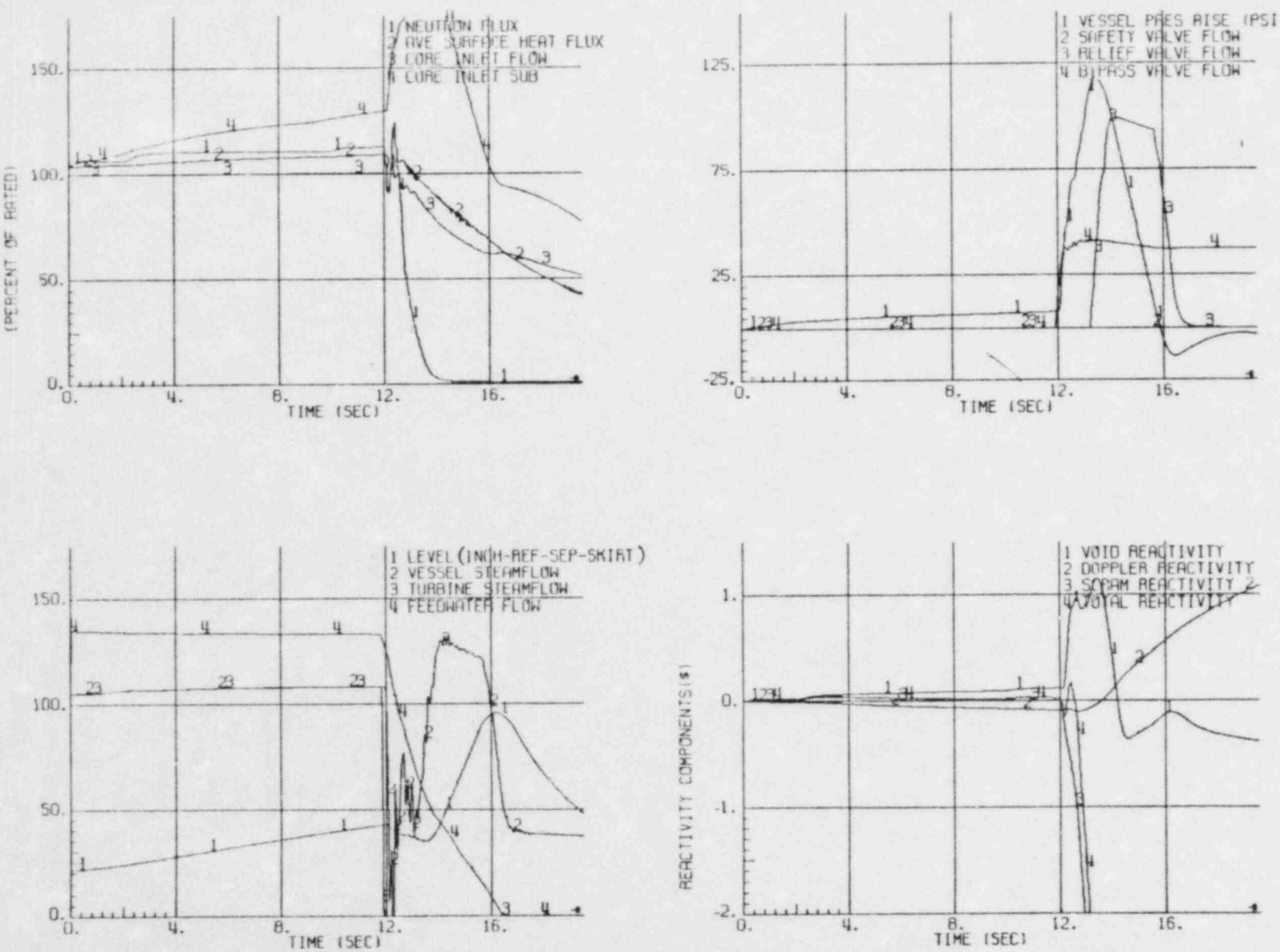

Figure 15.1-3. Feedwater Controller Failure, Maximum Demand, With High Water Level Trips

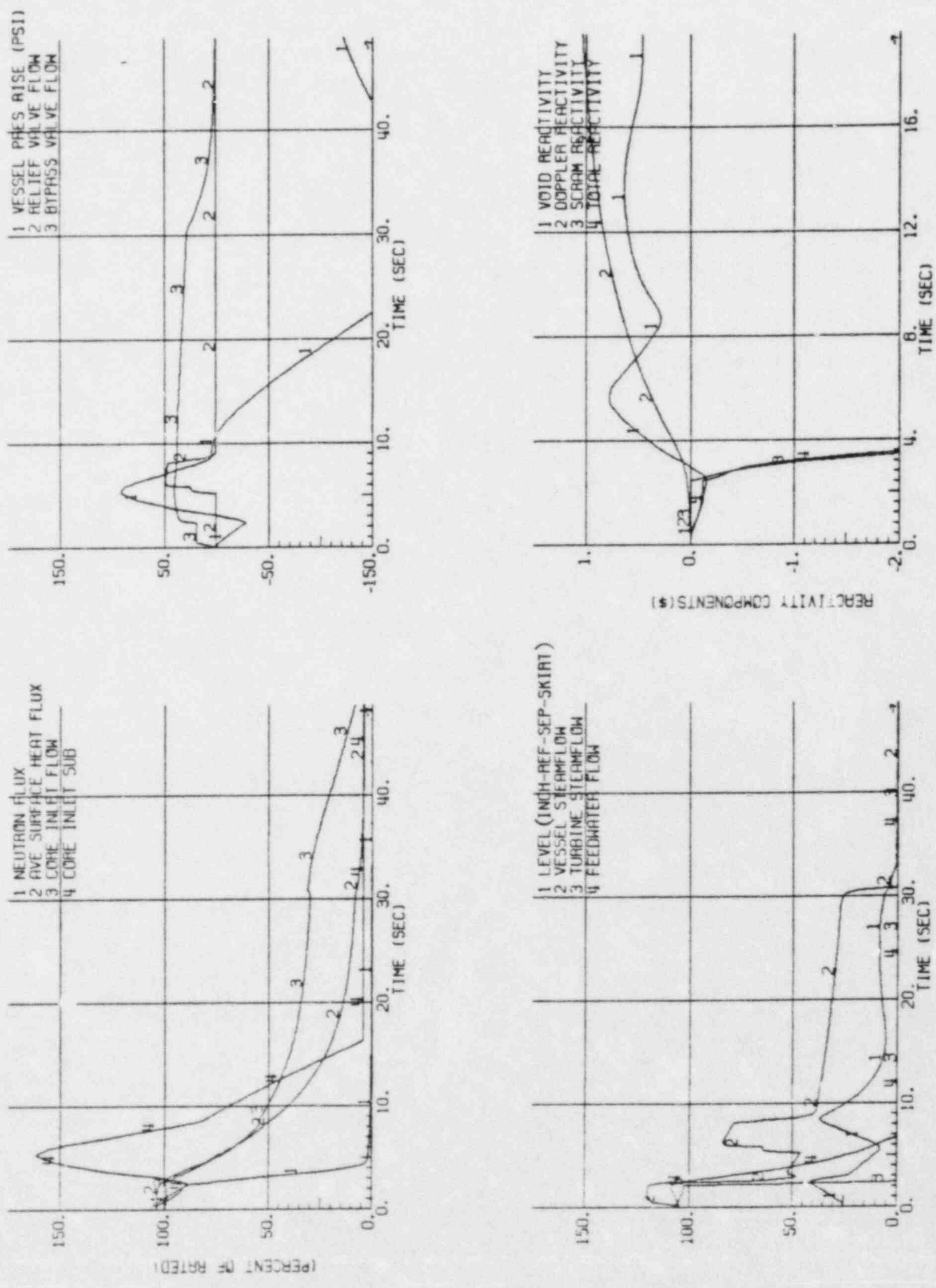

Figure 15.1-4. Pressure Regulator Failure Open to 130%

SECTION 15.2

TABLES

ILLUSTRATIONS

ILLUSTRATIONS (Continued)

SECTION 15.4

TABLES

SECTION 15.4

ILLUSTRATIONS

15.4.1.2.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, since no radioactive material is released from the fuel.

15.4.2 Rod Withdrawal Error at Power

15.4.2.1 Identification of Causes and Frequency Classification

15.4.2.1.1 Identification of Causes

The Rod Withdrawal Error (RWE) transient results from a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously until the Rod Withdrawal Limiter (RWL) function of the Rod Control and Information System (RCIS) blocks further withdrawal.

15.4.2.1.2 Frequency Classification

The frequency of occurrence for the RWE is assumed to be moderate, since definite data do not exist. The frequency of occurrence diminishes as the reactor approaches full power by virtue of the reduced number of control rod movements. A statistical approach, using appropriate conservative acceptance criteria, shows that consequences of the majority of RWEs would be very mild and hardly noticeable.

15.4.2.2 Sequence of Events and Systems Operation

15.4.2.2.1 Sequence of Events

The sequence of events for this transient is presented in Table 15.4-1.

15.4.2.2.2 System Operations

While operating in the power range in a normal mode of operation, the reactor operator makes a procedural error and withdraws the maximum worth control rod or gang of control rods continuously until the RWL inhibits further withdrawal. The RWL utilizes rod position indications of the selected rod as input.

During the course of this event, normal operation of plant instrumentation and conrols is assumed, although no credit is taken for this except as described above. No operation of any engineered safety feature (ESF) is required during this event.

15.4.2.3 Core and System Performance

15.4.2.3.1 Input Parameters and Initial Conditions]

The reactor core is assumed to be on MCPR and MLHGR technical specification limits prior to RWE initiation. A statistical analysis of the rod withdrawal error results (Appendix 15B) initiated from a wide range of operating conditions (exposure, power, flow, rod patterns, xenon conditions, etc) has been performed, establishing allowable rod withdrawal increments applicable to all BWR/6 plants. These rod withdrawal increments were determined such that the design basis ΔMCPR (minimum critical power ratio) for rod withdrawal errors initiated from the technical specification operating limit and mitigated by the RWL system withdrawal restrictions, provides a 95% probability at the 95% confidence level that any randomly occurring RWE will not result in a larger ΔMCPR. MCPR was verified to be the limiting thermal performance parameter and therefore was used to establish the allowable withdrawal increments. The 1% plastic strain limit on the clad was always a less limiting parameter.

15.4.4.2.1.1 Operator Actions

The normal sequence of operator actions expected in starting the idle loop is as follows. The operator should:

(1) adjust rod pattern, as necessary, for new power level following idle loop start;

(2) determine that the idle recirculation pump suction and discharge block values are open and that the flow control valve in the idle loop is at minimum position and, if not, place them in this configuration;

(3) readjust flow of the running loop downward to less than half of the rated flow;

(4) determine that the temperature difference between the two loops is no more than 50°F;

(5) start the idle loop pump and adjust flow to match the adjacent loop flow (monitor reactor power); and

(6) readjust power, as necessary, to satisfy plant requirements per standard procedure.

NOTE: The time to do the above work is approximately 1/2 hour.

15.4.4.2.2 Systems Operation

This event assumes and takes credit for normal functioning of plant instrumentation and controls. No protection systems action is anticipated. No ESF action occurs as a result of the transient.

15.4.4.3 Core and System Performance

15.4.4.3.1 Input Parameters and Initial Conditions

One recirculation loop is idle and filled with cold water (100°F). (Normal procedure when starting an idle loop with one pump already running requires that the indicated idle loop temperature be no more than 50°F lower than the indicated active loop temperature.)

The active recirculation loop is operating with the flow control valve position that produces about 70% of normal rated jet pump diffuser flow in the active jet pumps.

The core is receiving 33% of its normal rated flow. The remainder of the coolant flows in the reverse direction through the inactive jet pumps.

The idle recirculation pump suction and discharge block valves are open and the recirculation flow control valve is closed to its minimum open position. (Normal procedure requires leaving an idle loop in this condition to maintain the loop temperature within the required limits for restart.)

15.4.4.3.2 Results

The transient response to the incorrect startup of a cold, idle recirculation loop is shown in Figure 15.4-1. Shortly after the pump begins to move, a surge in flow from the started jet pump diffusers causes the core inlet flow to rise sharply. The motor approaches synchronous speed in approximately 3 sec because of the assumed minimum pump and motor inertia.

A short-duration neutron flux peak is produced as the colder, increasing core flow reduces the void volume. Surface heat flux follows the slower response of the fuel and peaks at 80% of rated

15.4.4.3.2 Results (Continued)

before decreasing after the cold water washed out of the loop at about 18 sec. No damage occurs to the fuel barrier and MCPR remains significantly above the safety limit as the reactor settles out at its new steady-state condition. Therefore, this event does not have to reanalyzed for specific core configurations.

15.4.4.4 Barrier Performance

No evaluation of barrier performance is required for this event since no significant pressure increases are incurred during this transient (Figure 15.4-1).

15.4.4.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, since no radioactive material is released from the fuel.

15.4.5 <u>Recirculation Flow Control Failure with Increasing Flow</u>

15.4.5.1 Identification of Causes and Frequency Classification

15.4.5.1.1 Identification of Causes

Failure of the master controller of neutron flux controller can cause an increase in the core coolant flow rate. Failure within a loop's flow controller can also cause an increase in core coolant flow rate.

15.4.5.1.2 Frequency Classification

This transient disturbance is classified as an incident of moderate frequency.

15.4.5.2 Sequence of Events and Systems Operation

15.4.5.2.1 Sequence of Events

15.4.5.2.1.1 Fast Opening of One Recirculation Valve

Table 15.4-4 lists the sequence of events for Figure 15.4-2.

15.4.5.2.1.2 Fast Opening of Two Recirculation Valves

Table 15.4-5 lists the sequence of events for Figure 15.4-3.

15.4.5.2.1.3 Identification of Operator Actions

Initial action by the operator should include:

(1) transfer flow control to manual and reduce flow to minimum, and

(2) identify cause of failure.

Reactor pressure will be controlled as required, depending on whether a restart or cooldown is planned. In general, the corrective action would be to hold reactor pressure and condenser vacuum for restart after the malfunctioning flow controller has been repaired. The following is the sequence of operator actions expected during the course of the event, assuming restart. The operator should:

(1) observe that all rods are in;

(2) check the reactor water level and maintain above low level (L2) trip to prevent MSLIVs from isolating;

(3) switch the reactor mode switch to the STARTUP position;

15.4.5.4 Barrier Performance

15.4.5.4.1 Fast Opening of One Recirculation Valve

This transient results in a very slight increase in reactor vessel pressure (Figure 15.4-2) and therefore represents no threat to the RCPB.

15.4.5.4.2 Fast Opening of Two Recirculation Valves

This transient results in a very slight increase in reactor vessel pressure (Figure 15.4-3) and therefore represents no threat to the RCPB.

15.4.5.5 Radiological Consequences

An evaluation of the radiological consequences is not required for this event, since no radioactive material is released from the fuel.

15.4.6 Chemical and Volume Control System Malfunctions

Not applicable to BWRs. This is a PWR event.

15.4.7 Misplaced Bundle Accident

15.4.7.1 Identification of Causes and Frequency Classification

15.4.7.1.1 Identification of Causes

The event discussed in this section is the improper loading of a fuel bundle and subsequent operation of the core. Three errors must occur for this event to take place in the equilibrium core loading. First, a bundle must be misloaded into a wrong location in the core. Second, the bundle which was supposed to be loaded where the mislocation occurred would have to also be put in an

15.4.7.1.1 Identification of Causes (Continued)

incorrect location or discharged. Third, the misplaced bundles would have to be overlooked during the core verification process performed following core loading.

15.4.7.1.2 Frequency Classification

This unlikely event occurs when a fuel bundle is loaded into the wrong location in the core. It is assumed the bundle is misplaced to the worst possible location, and the plant is operated with the mislocated bundle. This event is categorized as an infrequency incident based on the following data:

Expected Frequency: 0.002 events/operating cycle

The above number is based upon past experience.

15.4.7.2 Sequence of Events and Systems Operation

15.4.7.2.1 Sequence of Events

The postulated sequence of events for the misplaced bundle accident (MBA) is presented in Table 15.4-6.

15.4.7.2.2 Systems Operation

A fuel loading error, undetected by in-core instrumentation following fueling operations, may result in an undetected reduction in thermal margin during power operations. For the analysis reported herein, no credit for detection is taken and, therefore, no corrective operator action or automatic protection system functioning is assumed to occur.

Table 15.4-1

SEQUENCE OF EVENTS

Elapsed Time (sec)	
0	Core is operated in a typical control rod pattern on limits
0	Operator withdraws a single rod or gang of rods continuously
1	The local power in the vicinity of the withdrawn rod (or gang) increases. Gross core power increases.
$\sim$4	RWL blocks further withdrawal
$\sim$25	Core stabilizes at slightly higher core power level

*For a 1.0 ft RWL incremental withdrawal block. Time would be longer for a larger block, since rods are withdrawn at approximately 3 in./sec.

Table 15.4-2

ROD BLOCK ALARM DISTANCES (BWR/6)

Power Range (% of rated)	Allowable Withdrawal Distance (ft)
60 - 100	1.0
20 - 70	2.0
0 - 20	no restrictions*

*The BPWS function of the RCIS provides control of rod withdrawals below the 20% power setpoint and allows a maximum withdrawal distance of 9 ft.

Table 15.4-3

SEQUENCE OF EVENTS FOR FIGURE 15.4-1

Time (sec)	Event
0	Start pump motor
0.30	Jet pump diffuser flows on started pump side become positive
3.0	Pump motor at full speed and drive flow at about 21% of rated
18.0 (est)	Last of cold water leaves recirculation drive loop
18.1	Peak value of core inlet subcooling
50	Reactor variables settle into new steady state

Table 15.4-4

SEQUENCE OF EVENTS FOR FIGURE 15.4-2

Time (sec)	Event
0	Simulate failure of single loop control
1.3	Reactor APRM high-flux scram trip initiated
3.0 (est)	Turbine control valves start to close upon falling turbine pressure
6.5	Recirculation pump drive motors trip due to L3
25	Turbine control valves closed. Turbine pressure below pressure regulator setpoints
>100 (est)	Reactor variables settle into new steady-state

Table 15.4-5

SEQUENCE OF EVENTS FOR FIGURE 15.4-3

Time (sec)	Event
0	Initiate failure of master controller
1.6	Reactor APRM high-flux scram trip initiated
3.5 (est)	Turbine control valves start to close upon falling turbine pressure
5.6	Recirculation pump drive motors trip due to L3
32.0 (est)	Turbine control valves closed. Turbine pressure below pressure regulator setpoints
>100 (est)	Reactor variables settle into new steady-state

Table 15.4-6

SEQUENCE OF EVENTS FOR THE MISPLACED BUNDLE ACCIDENT

(1) During the core loading operation, a bundle is loaded into the wrong core location.

(2) Subsequently, the bundle designated for this location is incorrectly loaded into the location of the previous bundle.

(3) During the core verification procedure, the two errors are not observed.

(4) The plant is brought to full power operation without detecting misplaced bundle.

(5) The plant continues to operate throughout the cycle.

Table 15.4-7

INPUT PARAMETERS AND INITIAL CONDITIONS FOR THE FUEL BUNDLE LOADING ERROR

(1)	Power (% rated)	100
(2)	Flow (% rated)	100
(3)	MCPR operating limit*	1.20
(4)	MLHGR operating limit (kW/ft)*	13.4
(5)	Core Exposure	End of Cycle

*These are above the current operating limits. Since these limits do not go into the calculation of the MCPR associated with a mislocated bundle, differences in the safety operating limits will not effect these results.

Table 15.4-8

RESULTS OF MISPLACED BUNDLE ANALYSIS
EQUILIBRIUM CYCLE

(1)	MCPR Safety Limit	1.07
(2)	MCPR with misplaced bundle	1.14
(3)	LHGR 1% plastic strain limit	>20 kW/ft
(4)	LHGR with misplaced bundle*	14.9

*Does not include any densification penalty.

Table 15.4-9

SEQUENCE OF EVENTS FOR ROD DROP ACCIDENT

Approximate Elapsed Time (sec)	Event
	Reactor is operating at 50% rod density pattern.
	Maximum worth control rod blade becomes decoupled from the CRD.
	Operator selects and withdraws the control rod drive of the decoupled rod either individually or along with other control rods assigned to the RCIS group.
	Decoupled control rod sticks in the fully inserted or an intermediate bank position.
0	Control rod becomes unstuck and drops to the drive position at the nominal measured velocity plus three standard deviations.
<1	Reactor goes on a positive period and the initial power increase is terminated by the doppler coefficient.
<1	APRM 120% power signal scrams reactor.
<5	Scram terminates accident.

Table 15.4-10

INPUT PARAMETERS AND INITIAL CONDITIONS FOR ROD WORTH COMPLIANCE CALCULATION

1.	Reactor Power (% rated)	1
2.	Reactor Flow (% rated)	100
3.	Core Average Exposure (MWd/t)	Most reactive Point in cycle
4.	Control Rod Fraction	∿0.50
5.	Average Fuel Temperature (°C)	286
6.	Average Moderator Temperature (°C)	286
7.	Xenon State	None
8.	Core Average Void Fraction (%)	0

Table 15.4-11

INCREMENT WORTH OF THE MOST REACTIVE ROD USING BPWS

Core Condition	Control Rod Group	Banked At Notch	Control Rod (I,J)	Drops From-To	Increase (k_{eff})
3000	7	04	(26,35)	00-08	0.00248
3000	7	08	(26,35)	00-12	0.00278
3000	7	12	(26,35)	00-48	0.00269
3000	7	48	(26,35)	00-48	0.00198

NOTE: The following assumptions were made to ensure that the rod worths were conservatively high for the BPWS:

(a) BOC

(b) Hot Startup

(c) No Xenon

Table 15.4-12

CONTROL ROD DROP ACCIDENT EVALUATION PARAMETERS

	Design Basis Assumptions	Realistic Basis Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents:		
A. Power level	3651 MWt	3651 MWt
B. Burnup	NA	NA
C. Fuel damaged	770 rods	770 rods
D. Release of activity by nuclide	Table 15.4-14	Table 15.4-17
E. Iodine fractions:		
(1) Organic	0	0
(2) Elemental	1	1
(3) Particulate	0	0
F. Reactor coolant activity before the accident	NA	NA
G. Peaking factor	1.5	1.0
II. Data and assumptions used to estimate activity released:		
A. Condenser leak rate (%/day)	1.0	0.5
B. Turbine building leak rate (%/day)	NA	1327
C. Valve closure time (sec)	NA	5
D. Absorption and filtration efficiencies:		
(1) Organic iodine	NA	NA
(2) Elemental iodine	NA	NA
(3) Particulate iodine	NA	NA
(4) Particulate fission products	NA	NA
E. Recirculation system parameters:		
(1) Flow rate	NA	NA
(2) Mixing efficiency	NA	NA
(3) Filter efficiency	NA	NA
F. Containment spray parameters (flow rate, drop size, etc)	NA	NA
G. Containment volumes	NA	NA
H. All other pertinent data and assumptions	None	None

Table 15.4-12 (Continued)

		Design Basis Assumptions	Realistic Basis Assumptions
III.	Dispersion Data:		
	A. Site Boundary and LPZ distances (m)	*	*
	B. X/Q's for time intervals of:		
	(1) 0-1 hr - SB/LPZ	2.0E-3/1.0E-2	2.0E-3/1.0E-3
	(2) 1-8 hr - SB/LPZ	3.8E-4	3.8E-4
	(3) 8-16 hr - SB/LPZ	1.0E-4	1.0E-4
	(4) 16 hr-3 days - LPZ	3.4E-5	3.4E-5
	(5) 3-26 day - LPZ	7.5E-6	7.5E-6
IV.	Dose Data:		
	A. Method of dose calculation	Reference 3	Reference 4
	B. Dose conversion assumptions	Reference 3	Reference 4
	C. Peak activity concentrations in condenser	Table 15.4-13	Table 15.4-16
	D. Doses	Table 15.4-15	Table 15.4-18

*Applicant to Supply

Table 15.4-13

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY AIRBORNE IN CONDENSER (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.2E 03	2.1E 03	2.1E 03	2.0E 03	1.5E 03	1.2E 02
I132	3.6E 03	3.1E 03	2.7E 03	2.0E 03	1.1E 03	3.2E 02	9.4E 01	2.5E 00	7.5E-10	0.
I133	3.3E 03	3.3E 03	3.2E 03	3.1E 03	2.9E 03	2.6E 03	2.2E 03	1.5E 03	1.3E 02	9.5E-08
I134	5.6E 03	3.8E 03	2.6E 03	1.2E 03	2.4E 02	1.0E 01	4.2E-01	3.1E-05	0.	0.
I135	4.7E 03	4.5E 03	4.2E 03	3.8E 03	3.1E 03	2.0E 03	1.3E 03	3.7E 02	1.8E-01	0.
Total I	1.9E 04	1.7E 04	1.5E 04	1.2E 04	9.5E 03	7.0E 03	5.7E 03	3.9E 03	1.6E 03	1.2E 02
Kr83m	2.5E 04	2.1E 04	1.8E 04	1.2E 04	5.7E 03	1.3E 03	2.8E 02	3.2E 00	5.8E-12	0.
Kr85m	6.1E 04	5.6E 04	5.2E 04	4.5E 04	3.3E 04	1.8E 04	9.5E 03	1.5E 03	2.0E-02	0.
Kr85	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.6E 03	1.5E 03	1.5E 03	1.2E 03
Kr87	1.2E 05	9.5E 04	7.3E 04	4.2E 04	1.4E 04	1.6E 03	1.8E 02	2.5E 01	0.	01
Kr88	1.8E 05	1.6E 05	1.4E 05	1.1E 05	6.6E 04	2.4E 04	9.1E 03	4.6E 02	7,8E-06	0.
Kr89	1.8E 05	3.1E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.4E 03	1.2E 03	2.0E 02
Xe131m	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.5E 03	1.4E 03	1.2E 03	2.0E 02
Xe133m	6.1E 04	6.1E 04	6.0E 04	6.0E 04	5.8E 04	5.5E 04	5.2E 04	4.4E 04	1.7E 04	4.1E 00
Xe133	3.6E 05	3.5E 05	3.5E 05	3.5E 05	3.5E 05	3.4E 05	3.3E 05	3.1E 05	2.0E 05	5.1E 03
Xe135m	9.7E 04	2.6E 04	6.7E 03	4.4E 02	1.9E 00	3.6E-05	6.9E-10	9.	0.	0.
Xe135	6.5E 04	6.2E 04	6.0E 04	5.6E 04	4.8E 04	3.5E 04	2.6E 04	1.0E 04	4.3E 01	0.
Xe137	3.9E 05	2.1E 03	9.2E 00	1.8E-04	7.0E-14	0.	0.	0.	0.	0.
Xe138	4.3E 05	1.0E 05	2.4E 04	1.3E 03	3.6E 00	2.9E-05	2.4E-10	0.	0.	01
Total NG	2.0E 06	9.4E 05	7.9E 05	6.8E 05	5.7E 05	4.8E 05	4.3E 05	3.7E 05	2.2E 05	6.5E 03

Table 15.4-14

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
ACTIVITY RELEASED TO ENVIRONMENT (Ci)

Isotope	1 min	30 min	1 hr	2 hr	4 hr	8 hr	12 hr	1 day	4 day	30 day
I131	1.5E-02	4.6E-01	9.1E-01	1.8E 00	3.6E 00	7.2E 00	1.1E 01	2.1E 01	7.3E 01	2.1E 02
I132	2.5E-02	7.0E-01	1.3E 00	2.3E 00	3.5E 00	4.5E 00	4.8E 00	4.9E 00	4.9E 00	4.9E 00
I133	2.3E-02	6.9E-01	1.4E 00	2.7E 00	5.2E 00	9.8E 00	1.4E 01	2.3E 01	4.0E 01	4.1E 01
I134	3.9E-02	0.8E-01	1.6E 00	2.4E 00	2.9E 00	3.0E 00	3.0E 00	3.0E 00	3.0E 00	3.0E 00
I135	3.3E-02	9.5E-01	1.9E 00	3.5E 00	6.4E 00	1.1E 01	1.3E 01	1.7E 01	1.9E 01	1.9E 01
Total I	1.4E-01	3.8E 00	7.1E 00	1.3E 01	2.2E 01	3.5E 01	4.6E 01	6.9E 01	1.4E 02	2.8E 02
Kr83m	1.8E-01	4.9E 00	8.9E 00	1.5E 01	2.2E 01	2.7E 01	2.8E 01	2.8E 01	2.8E 01	2.8E 01
Kr85m	4.2E-01	1.2E 01	2.4E 01	4.4E 01	2.7E 00	1.2E 00	1.4E 02	1.6E 02	1.6E 02	1.6E 02
Kr85	1.1E-02	3.3E-01	6.5E-01	1.3E 00	2.6E 00	5.2E 00	7,8E 00	1.6E 01	6.1E 01	4.0E 02
Kr87	8.7E-01	2.3E 01	4.0E 01	6.4E 01	8.5E 01	9.4E 01	9.5E 01	9.5E 01	9.5E 01	9.5E 01
Kr88	1.2E 00	3.5E 01	6.6E 01	1.2E 02	1.9E 02	2.6E 02	2.8E 02	3.0E 02	3.0E 02	3.0E 02
Kr89	1.4E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00	7.1E 00
Xe131m	1.1E-02	3.2E-01	6.3E-01	1.3E 00	2.5E 00	5.0E 00	7.5E 00	1.5E 01	5.4E 01	2.0E 02
Xe133m	4.3E-01	1.3E 01	2.5E 01	5.0E 01	9.9E 01	1.9E 02	2.8E 02	5.2E 02	1.4E 03	1.9E 03
Xe133	2.5E 00	7.4E 01	1.5E 02	2.9E 02	5.9E 02	1.2E 03	1.7E 03	3.3E 03	1.1E 04	2.5E 04
Xe135m	6.9E -01	1.2E 01	1.4E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01	1.5E 01
Xe135	4.5E-01	1.3E 01	2.6E 01	5.0E 01	9.3E 01	1.6E 02	2.1E 02	3.0E 02	3.5E 02	3.5E 02
Xe137	3.0E 00	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01	1.8E 01
Xe138	3.0E 00	4.9E 01	6.0E 01	6.3E 01	6.4E 01	6.4E 01	6.4E 01	6.4E 01	6.4E 01	6.4E 01
Total NG	1.4E 01	2.6E 02	4.4E 02	7.4E 02	1.3E 03	2.1E 03	2.9E 03	4.9E 03	1.3E 04	2.8E 04

TABLE 15.4-15

CONTROL ROD DROP ACCIDENT
(DESIGN BASIS ANALYSIS)
Radiological Effects

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area	0.22	2.55
Low Population Zone	0.16	4.08

Table 51.4-16

CONTROL ROD DROP ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY AIRBORNE IN THE CONDENSER (Ci)

Isotope	1 min	1 hr	2 hrs	8 hrs	1 day	4 days	30 days
I131	2.92E-01	2.91E-01	2.90E-01	2.84E-01	2.68E-01	2.07E-01	2.20E-02
I132	4.47E-02	3.31E-02	2.45E-02	3.96E-03	3.08E-05	9.65E-15	0.
I133	1.43E-01	1.38E-01	1.34E-01	1.09E-01	6.42E-02	5.80E-03	5.46E-12
I134	3.36E-02	1.54E-02	6.99E-03	6.04E-05	1.89E-10	0.	0.
I135	1.09E-01	9.84E-02	8.86E-02	4.72E-02	8.77E-03	4.48E-06	0.
Total	6.23E-01	5.77E-01	5.44E-01	4.45E-01	3.41E-01	2.13E-01	2.20E-02
Kr83m	3.35E 01	2.32E 01	1.59E 01	1.68E 00	4.20E-03	7.78E-15	0.
Kr85m	2.28E 02	1.95E 02	1.67E 02	6.60E 01	5.53E 00	7.73E-05	0.
Kr85	2.26E 02	2.26E 02	2.26E 02	2.26E 02	2.25E 02	2.21E 02	1.94E 02
Kr87	1.91E 02	1.12E 02	6.46E 01	2.42E 00	3.80E-04	0.	0.
Kr88	4.30E 02	3.37E 02	2.63E 02	5.94E 01	1.13E 00	1.94E-08	0.
Kr89	1.13E-01	2.67E-07	5.06E-13	0.	0.	0.	0.
Xe131m	2.87E 01	2.86E 01	2.85E 01	2.81E 01	2.69E 01	2.23E 01	4.36E 00
Xe133m	4.41E 02	4.35E 02	4.29E 02	3.97E 02	3.21E 02	1.24E 02	3.39E-02
Xe133	4.27E 03	4.25E 03	4.22E 03	4.08E 03	3.73E 03	2.47E 03	7.17E 01
Xe135m	3.23E 00	2.23E-01	1.47E-02	1.22E-09	9.	0.	0.
Xe135	8.99E 02	8.34E 02	7.73E 02	4.91E 02	1.46E 02	6.17E-01	0.
Xe137	4.01E-01	9.45E-06	1.86E-10	0.	0.	0.	0.
Xe138	6.17E 01	3.46E 00	1.84E-01	4.26E-09	0.	0.	0.
Total	6.81E 03	6.44E 03	6.19E 03	5.35E 03	4.45E 03	2.84E 03	2.70E 02

Table 15.4-17

CONTROL ROD DROP ACCIDENT
(REALISTIC ANALYSIS)
ACTIVITY RELEASED TO THE ENVIRONMENT (Ci)

Isotope	1 min	1 hr	2 hrs	8 hrs	1 day	4 days	30 days
I131	4.66E-09	1.41E-05	4.79E-05	3.72E-04	1.29E-03	4.84E-03	1.55E-02
I132	7.14E-10	1.78E-06	5.07E-06	1.72E-05	1.99E-05	1.99E-05	1.99E-05
I133	2.28E-09	6.75E-06	2.25E-05	1.59E-04	4.41E-04	8.05E-04	8.42E-04
I134	5.39E-10	1.00E-06	2.21E-06	3.67E-06	3.69E-06	3.69E-06	3.69E-06
I135	1.74E-09	4.93E-06	1.58E-05	8.88E-05	1.65E-04	1.82E-04	1.82E-04
Total	9.94E-09	2.86E-05	9.34E-05	6.41E-04	1.92E-03	5.85E-03	1.66E-02
Kr83m	5.35E-07	1.28E-03	3.50E-03	1.02E-02	1.12E-02	1.12E-02	1.12E-02
Kr85m	3.63E-06	9.97E-03	3.09E-02	1.51E-01	2.32E-01	2.40E-01	2.40E-01
Kr85	3.61E-06	1.09E-02	3.72E-02	2.92E-01	1.04E 00	4.40E 00	3.13E 01
Kr87	3.06E-06	6.59E-03	1.64E-02	3.60E-02	3.69E-02	3.69E-02	3.69E-02
Kr88	6.86E-06	1.78E-02	5.21E-02	2.01E-01	2.50E-01	2.50E-01	2.50E-01
Kr89	1.94E-09	8.96E-08	8.96E-08	8.96E-08	8.96E-08	8.96E-08	8.96E-08
Xe131m	4.57E-07	1.38E-03	4.70E-03	3.67E-02	1.28E-01	4.97E-01	1.93E 00
Xe133m	7.03E-06	2.11E-02	7.14E-02	5.37E-01	1.73E 00	4.85E 00	6.82E 00
Xe133	6.81E-05	2.06E-01	6.98E-01	5.39E 00	1.84E 01	6.43E 01	1.52E 02
Xe135m	5.23E-08	3.49E-05	4.30E-05	4.38E-05	4.38E-05	4.38E-05	4.38E-05
Xe135	1.43E-05	4.14E-02	1.35E-01	8.29E-01	1.77E 00	2.17E 00	2.18E 00
Xe137	6.80E-09	4.48E-07	4.48E-07	4.48E-07	4.48E-07	4.48E-07	4.48E-07
Xe138	1.00E-06	6.04E-04	7.21E-04	7.31E-04	7.31E-04	7.31E-04	7.31E-04
Total	1.09E-04	3.17E-01	1.05E 00	7.48E 00	2.36E 01	7.68E 01	1.95E 02

TABLE 15.4-18

CONTROL ROD DROP ACCIDENT
(REALISTIC ANALYSIS)
RADIOLOGICAL EFFECTS

	Whole Body Dose (rem)	Inhalation Dose (rem)
Exclusion Area	9.4E-05	5.4E-05
Low Population Zone	1.7E-04	2.0E-04

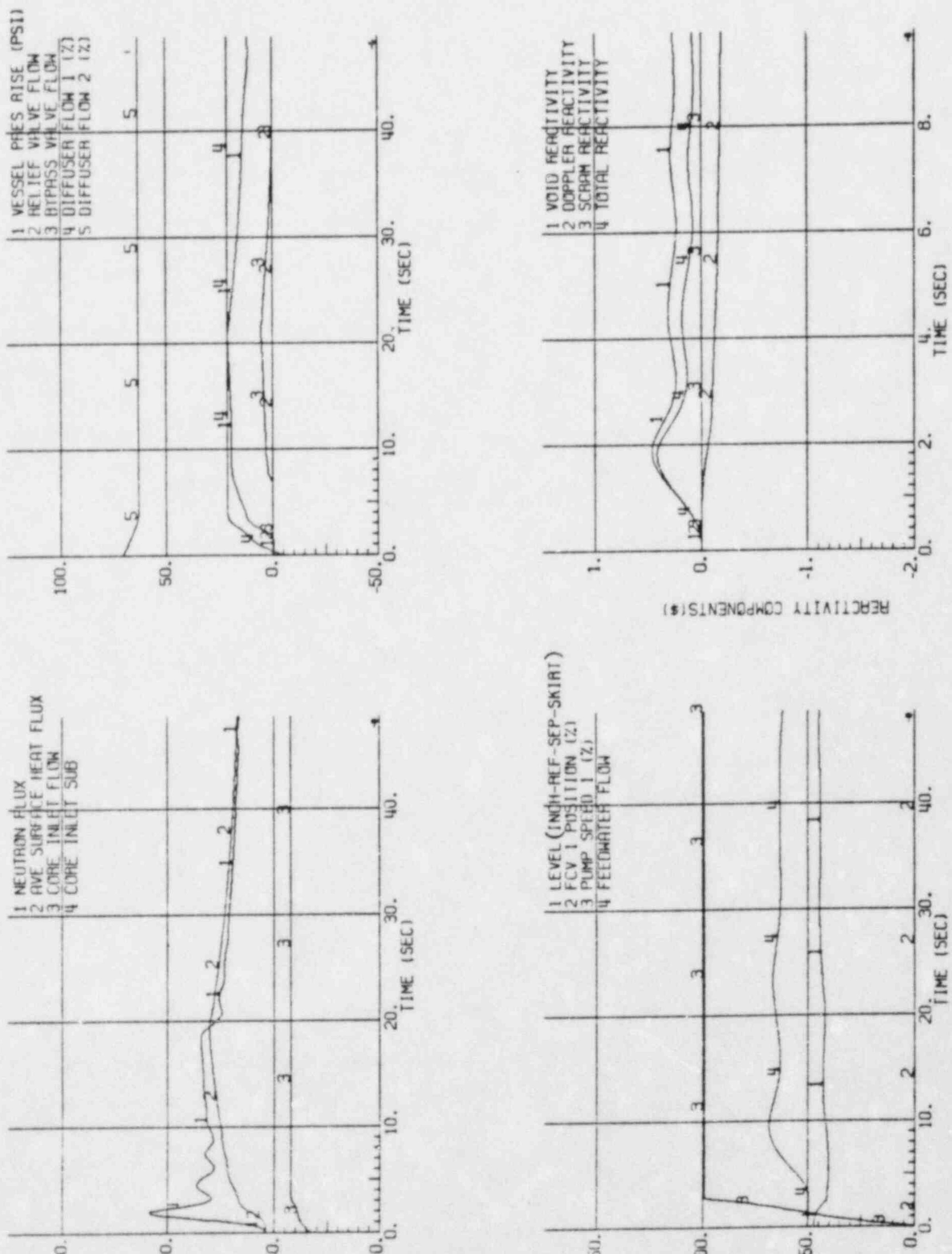

Figure 15.4-1. Startup of Idle Recirculation Loop Pump

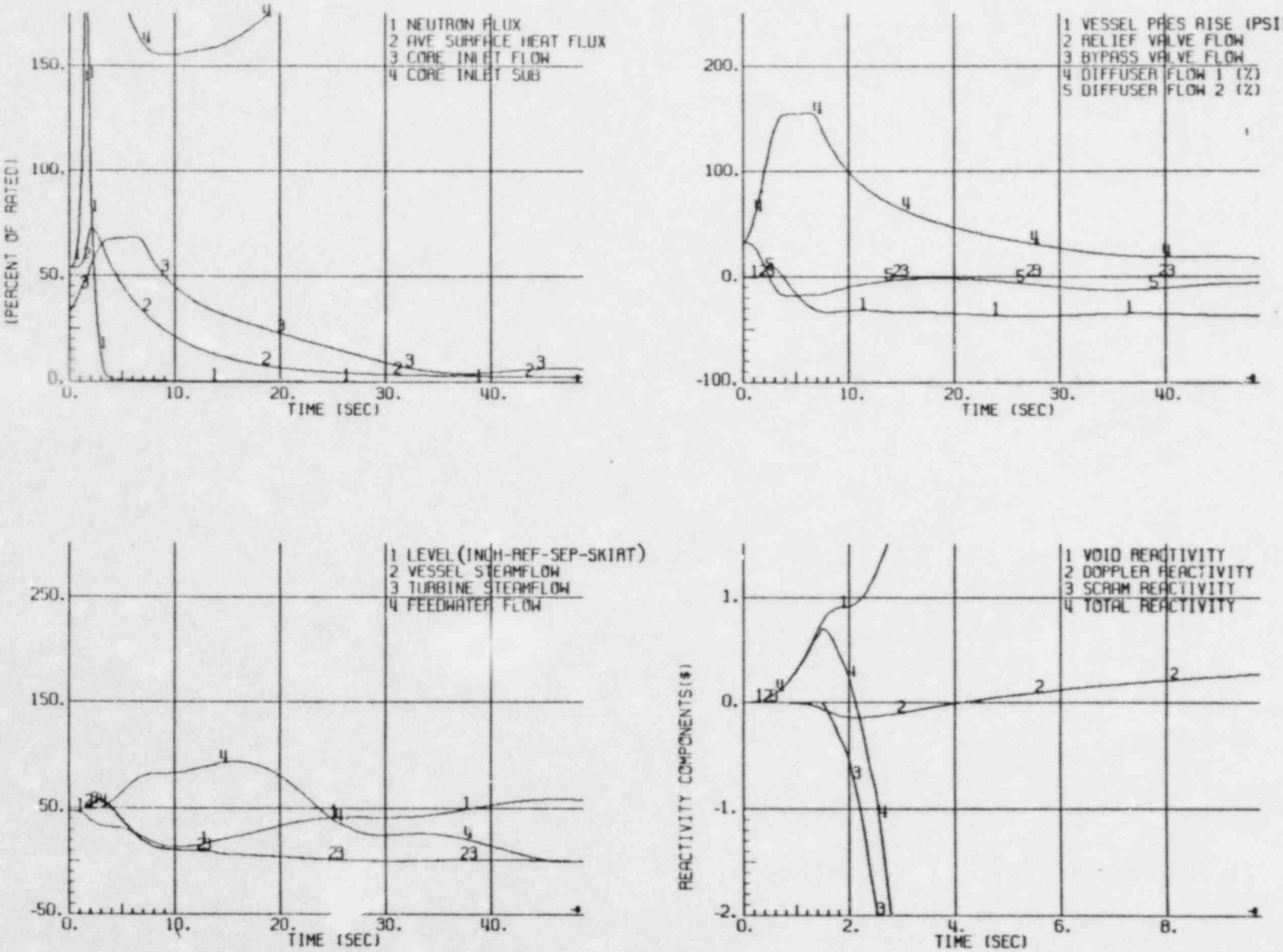

Figure 15.4-2. Fast Opening of One Recirc Valve at 30%/sec

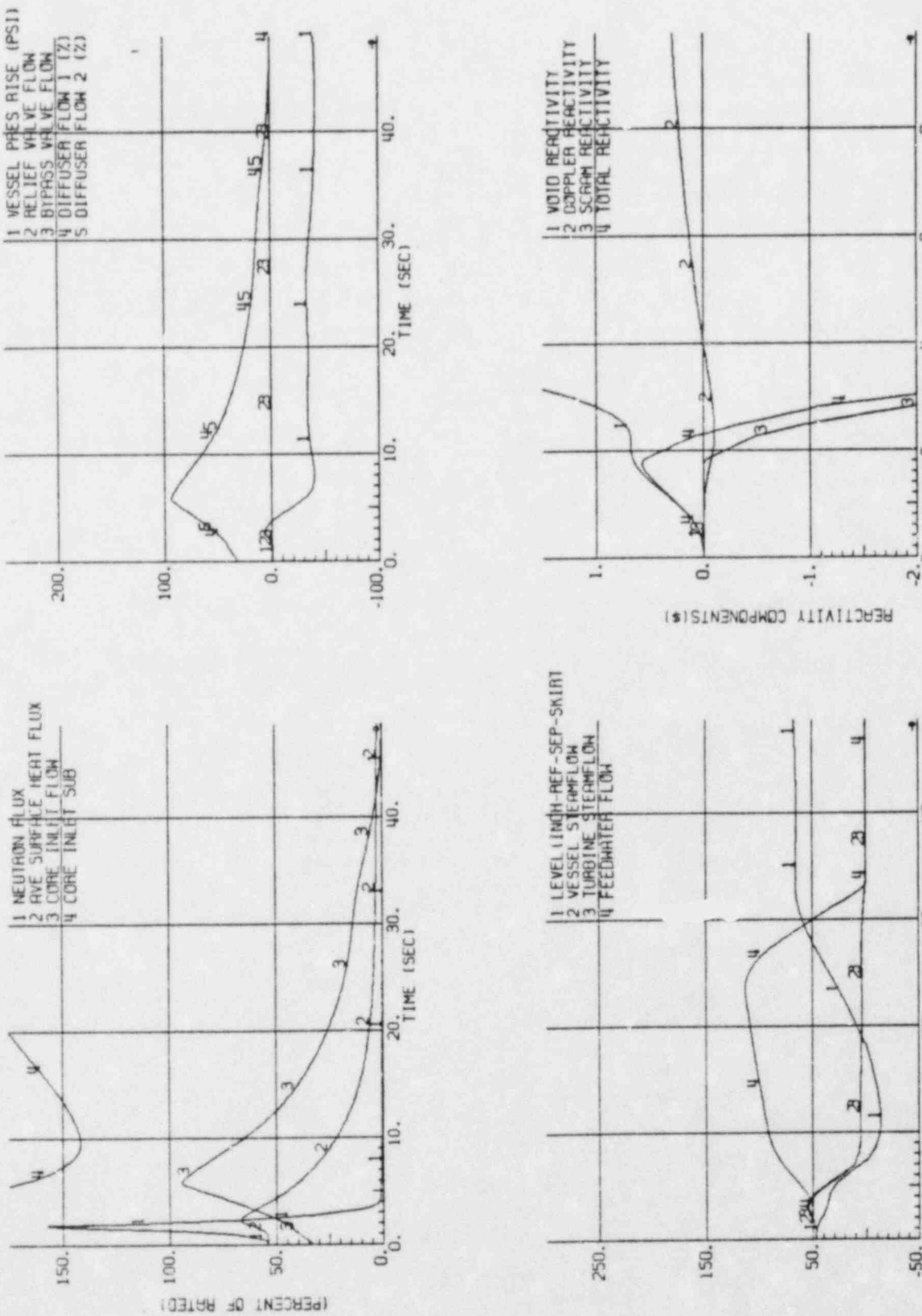

Figure 15.4-3. Fast Opening of Both Recirc Valves at 11%/sec

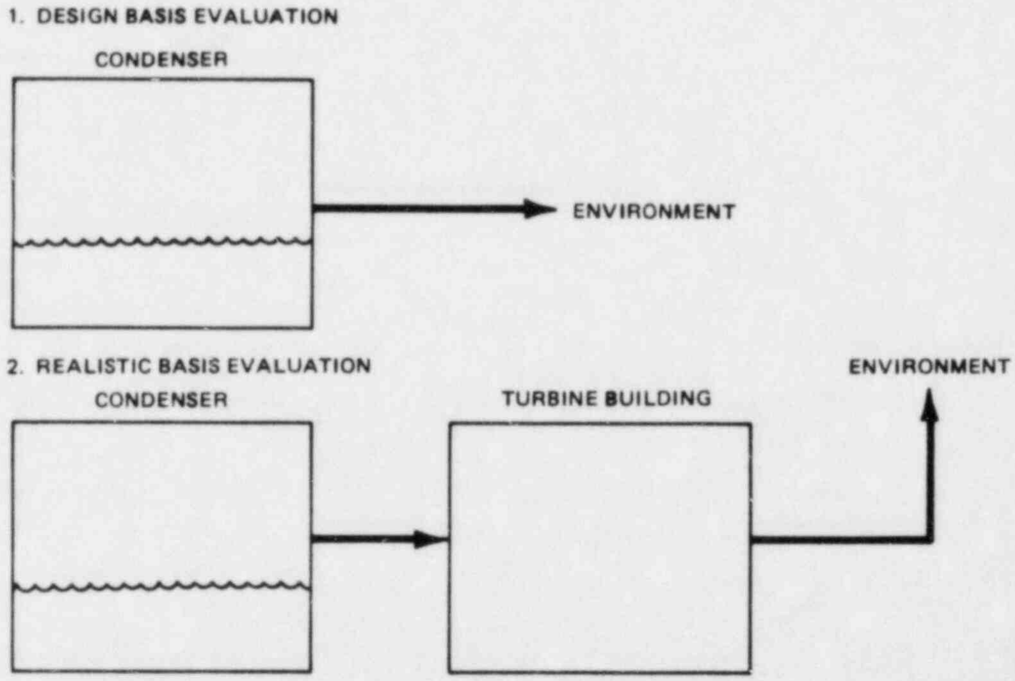

Figure 15.4-4. Leakage Path Model for Rod Drop Accident

15D.2 BWR PREVENTION AND MITIGATION CAPABILITY

This section describes the basic features and capabilities of the BWR/6 238 Nuclear Island which prevent or mitigate severe accidents.

The 238 Nuclear Island includes design features which prevent damage to the reactor core from transients or accidents (Subsection 15D.2.1). In the extremely unlikely event that extensive multiple failures result in core damage, additional features are provided to mitigate the effects of those accidents (Subsection 15D.2.2). The adequacy of this design is quantified by analysis of the probability of core damage and the risk to the public. These evaluations are discussed in the probabilistic risk assessment presented in Section 15D.3.

In addition to the design features and capabilities, the development of procedural guidelines to be used by operators in the event of an accident are made part of the standard design process. A discussion of the emergency procedure guidelines is provided in Subsection 15D.2.3.

15D.2.1 BWR Prevention Features and Capabilities

The BWR safety approach has traditionally stressed the prevention of core damage as a key objective of the design to ensure plant safety. By providing multiple and diverse methods of water injection to the reactor vessel, the likelihood of core damage due to inadequate core cooling is minimized. Subsection 15D.3.3, Probability of Core Damage, provides quantitative results to support the high degree of success of this safety approach.

15D.2.1 BWR Prevention Features and Capabilities (Continued)

The basic features and capability of the 238 Nuclear Island are described in GESSAR Chapters 5, 6 and 7. Following the accident at Three Mile Island, several design changes were identified by the NRC and the Nuclear Industry which resulted in relatively minor changes to the basic design described in Chapters 5, 6 and 7. Some of these changes (Subsection 15D.2.1.3) have an effect on the probabilistic risk assessment discussed in Section 15D.3. Other changes, though not quantifiable in terms suitable for inclusion in a PRA, provide an additional positive improvement to plant safety. Changes in response to all applicable post-TMI requirements are described in Appendix 1A.

The principal BWR core protection functions are provided by 1) the reactor protection system, 2) the systems to supply water to the reactor core to provide adequate core cooling and 3) the systems to remove the decay heat. These functions are described below.

15D.2.1.1 Reactor Protection

The reactor protection system is described in Chapter 7 (Section 7.2). The reactor protection function is performed by a control rod drive system (Section 4.6) and the Standby Liquid Control System (Section 7.4). These systems provide reliable and diverse methods of controlling reactor neutron flux and achieving reactivity control when protective action is required.

15D.2.1.2 Core Cooling

The ability of the BWR plant to supply water to the core following transients and accidents is provided by a combination of high pressure and low pressure water delivery

15D.2.1.2 Core Cooling (Continued)

systems as listed in Figure 15D.2-1. The means to rapidly depressurize the reactor vessel in the event that high pressure systems are unavailable is also provided.

The high pressure systems consist of the main feedwater system, the High Pressure Core Spray (HPCS) system (Section 6.3), the Reactor Core Isolation Cooling (RCIC) system (Subsection 5.4.6) and flow from the Control Rod Drive (CRD) Hydraulic system.

The low pressure injection systems consist of three independent low pressure coolant injection (LPCI) loops which are part of the Residual Heat Removal (RHR) system, the Low Pressure Core Spray system (LPCS), and the condensate system. The LPCI and LPCS systems are described in Section 6.3. The condensate system can provide makeup water to the reactor at low pressure independent of the availability of the feedwater system.

The capability to rapidly depressurize the pressure vessel is provided by the Automatic Depressurization System (ADS) which is described in Section 7.3. In addition to this automatic system, the operator can manually depressurize the pressure vessel through the Safety Relief Valves or by using the main condenser as a heat sink.

Several key protection functions result from the ability to depressurize the BWR. If the high pressure makeup systems should all be unavailable, actuation of the ADS will depressurize the reactor vessel so that one or more of the low pressure systems may be used to maintain water level. When depressurized, all water sources are available for injection into the reactor vessel. These sources include the suppression

15D.2.1.2 Core Cooling (Continued)

pool, the condensate storage tank, main condensers and the service water supply. Depressurization also causes an increase in the actual water level in the core through creation of voids. Finally, RPV depressurization reduces any dynamic loads which the containment may be experiencing. These features provide a unidirectional operator action (depressurization) in response to core inventory threatening situations which facilitates the development of the symptom-based emergency procedure guidelines. (Subsection 15D.2.3)]

Unidirectional operator action in response to events is a key feature of the BWR design. Although water delivery and depressurization systems perform automatically, the operator is not burdened with decisions as to the correct course of action to assure adequate core cooling. In general, these actions are to maintain RPV water level in the normal range, and, in the extreme, depressurize the reactor vessel and ensure at least one water delivery system is operating. Subsection 15D.2.3 provides further detail on the development and content of the Emergency Procedure Guidelines.

The diversity of water delivery systems is a key feature of the BWR design which assures prevention of core damage following accidents or transients. Diversity is also found on the power supplies for the equipment, the motive force for components and in the method of water delivery.

Diversity in power supply to High Pressure and Low Pressure ECCS equipment is provided through use of different diesel generator vendors for the division 3 (HPCS) power as compared with division 1 or 2 (LPCI, LPCS) power. Direct-current powered valves and controls are supplied by batteries and alternating-current powered inverters. The motive forces used to drive high pressure water delivery systems are

15D.2.1.2 Core Cooling (Continued)

diverse through use of a steam turbine for RCIC and Diesel Electric drive for HPCS. Water delivery is provided through flooding (delivery to the core shroud with flow up through the bottom of the core) and spray (delivery through spray nozzles above the top of active fuel).

These diverse features of the system design add to the reliability of the systems and contribute to the low likelihood of core damage.

The number of pumps and pump flow capability associated with high pressure and low pressure injection systems are summarized in Table 15D.2-1. The plant response to design basis accidents is provided in Chapter 15. These analyses only take credit for the emergency core cooling system (ECCS). The Chapter 15 accident analyses show that the plant response is well within the licensing basis of the plant. The benefit of the BWR single vessel design is that at least seven pumps] in addition to the ECCS and RCIC pumps have the capability to inject directly to the reactor vessel and maintain the water level even if none of the ECCS pumps are available.

The probabilistic risk assessment in Subsection 15D.3.3 provides success criteria for adequate core cooling during transients and accidents which have degraded far beyond design basis assumptions. These criteria are based on realistic calculations of not exceeding 2200°F peak clad temperature. These criteria show the capability of the high pressure and low pressure water delivery systems to provide adequate core cooling for all initiating events including the large loss-of-coolant accident.

Indication of successful core cooling is provided to the operators in the control room by the reactor pressure vessel

15D.2.1.2 Core Cooling (Continued)

the measurement concept). This system, which is described further in Section 7.7, consists of redundant condensing chambers and differential pressure transmitters. By providing an indication of water level inventory in the vessel above the reactor core, and by providing an indication of the trend of the water level, the operator is assured that water is available to adequately cool the core. As backup to the trend indication of water level, flow indication of the high pressure and low pressure systems which supply water to the core is also provided. If there is an indication that the water level is dropping below the normal range, the operator has a single clear and direct action to assure continued adequate core cooling: increase the water level.

15D.2.1.3 Decay Heat Removal

15D.2.1.3.1 Vessel Heat Removal

Removal of the reactor decay heat from the pressure vessel is accomplished via release through the main steam system to the main condenser, through the RHR heat exchangers in either the steam condensing mode or shutdown cooling mode, or through Safety/Relief valves to the suppression pool. These heat exchangers also are capable of removing the energy stored in the suppression pool.

During an accident or transient, if the main steam line is isolated, or if a momentary pressure increase occurs, decay heat removal from the core is accomplished by discharge of steam to the suppression pool through Safety/ Relief valves (shown simplistically in Figure 15D.2-2). During relief valve action, the strong natural circulation down the core shroud and up through the core and steam separators ensures a passive means of decay heat removal within the reactor vessel with no reliance on active systems external to the

15D.2.1.3.1 Vessel Heat Removal (Continued)

reactor vessel. Accumulation of non-condensible gases is minimized by relief valve operation. Further, non condensible gases in the top of the vessel do not affect the natural circulation flow.]

If the main steam line is not isolated, core decay heat removal can be achieved via the Main Steam System provided a condenser vacuum can be maintained and a return flow path to the reactor vessel can be established through the condensate pumps, condensate booster pumps and feedwater pumps. This mode of operation requires the availability of offsite power, but is the normal means of decay heat removal. Decay heat is removed from the core by natural circulation as previously described for conditions of main steam line isolation.

Finally, when the reactor has been depressurized below 150 psia, the shutdown cooling mode of the RHR system may be used (Subsection 5.4.7). This system is operable on either onsite or offsite power and provides long-term core cooling by providing flow directly to and from the reactor vessel.

The suppression pool acts as a passive heat sink to absorb the decay heat release for many hours after a plant transient or accident without the need for operator action. The passive heat sink capability of the Mark III suppression pool is summarized in Table 15D.2-2 and shown in Figure 15D.2-4. These data are based on the initial condition of a design-basis loss of coolant accident (DBA) except that no active suppression pool cooling is assumed. Because the calculation continues well beyond the initial blowdown resulting from the DBA, the results are equally applicable to a less severe transient which results in a blowdown to the suppression

15D.2.1.3.1 Vessel Heat Removal (Continued)

pool and the maintenance of low RPV pressure for a long period of time. While still conservative, the results differ from other evaluations of containment pressure provided in Subsection 6.2.1 and 15D.3, Appendix F, due to more realistic assumptions on the relation between containment airspace temperature and suppression pool water temperature.

As shown in Figure 15D.2-4, the initial blowdown energy (consisting of stored vessel heat, 120 seconds of rated feedwater flow, vessel sensible heat, and 200 seconds of decay heat) increases the suppression pool temperature by about 60°F and causes little change in the containment (wetwell) air space pressure (after vacuum breakers have equalized drywell and containment pressures). As decay heat continues to be discharged to the suppression pool through the safety relief valves, the suppression pool temperature increases to a point where boiling of the suppression pool may begin. At this point, slightly more than twice the initial blowdown energy has been added to the containment.

After suppression pool boiling begins, the decay heat is transferred to the containment air space and structures. As shown on Figure 15D.2-4, the containment design pressure of 30 psia is not reached for about 27 hours.

At this point about four times the initial blowdown energy has been transferred to the containment. The containment ultimate pressure capability extends well beyond its design pressure as discussed in Section 15D.3, Appendix G. Figure 15D.2-4 shows that this point is not reached for about 48 hours after initiation of an event in which no active heat removal from the containment occurs. Under these

15D.2.1.4.1 Automatic RCIC Restart (Continued)

availability of the RCIC system during transients and accidents by allowing the system to automatically restart following high vessel water level shutoff.

Existing System Operation

The RCIC system is described in Subsection 5.4.6. During normal plant operation the steam supply valve to the turbine is closed. Upon receipt of a vessel low water level signal, the RCIC system starts automatically. The following automatic actions occur:

1. The steam supply valve to the turbine opens to supply steam to the turbine. Steam line drain isolation valves then close, which isolates the RCIC steam supply from the main condenser.

2. Once the steam supply valve leaves the fully closed position the ramp generator "ramp" function is initiated. This ramp generator controls the acceleration of the turbine via the turbine control valve.

3. The gland seal system automatically starts.

4. Condensate suction valve remains open or is automatically opened to supply water to the RCIC pump.

5. The pump discharge valve opens to supply the water to the reactor vessel.

6. The cooling water supply valve opens automatically and coolant is supplied to the turbine lube oil cooler.

15D.2.1.4.1 Automatic RCIC Restart (Continued)

7. The test bypass valve to the condensate storage tank closes, if initially open.

The RCIC system will automatically shut down upon receipt of any of the following signals:

1. Reactor high water level (see modification below)
2. RCIC pump low suction pressure
3. Turbine high exhaust pressure
4 Turbine overspeed
5. Auto-isolation signal
6. Manual turbine trip pushbutton

The shutdown is affected by releasing the spring-loaded turbine trip valve. In order to reset the system it is necessary to first close the steam supply valve, then drive the motor operator of the turbine trip valve in the close direction until the spring-loaded closing latch mechanism is reset. Finally, the turbine trip valve is driven to the full open position. Closure of the steam supply valve also resets the ramp generator, closes the vessel injection valve, closes the minimum flow valve and opens the appropriate drain valves.

Automatic Reset Modification

The planned change (Figure 15D.2-5) utilizes the steam supply valve to shut off steam to the turbine following reactor high water level, rather than using the turbine trip valve. Closure of the steam supply valve puts the system in a partial standby configuration because of the existing interlocks associated with closure of this valve. This plant change will be reflected in Subsection 5.4.6 following staff approval.

15D.2.1.4.1 Automatic RCIC Restart (Continued)

Effect of the Planned Changes

The planned change will utilize the RCIC steam supply valve (E51-F045) to shut off the steam to the turbine on high vessel level rather than the turbine trip valve. The steam supply valve will now be used to both initiate system operation at low reactor vessel water level and terminate system operation at high water level.

The time taken to shut off steam flow will be longer due to the nominally longer travel time of the steam supply valve compared to the trip valve. The spring-loaded turbine trip valve closes essentially instantaneously. The steam supply valve closes in fifteen seconds or less. Conservatively assuming full rated flow throughout this extended shutoff period and a maximum rated RCIC flow of 800 gpm, an additional 200 gallons will be added to the reactor vessel following the high vessel water level trip. This volume addition has an insignificant effect on high vessel level transients including those involving high-flow rate systems (e.g., HPCS) (Subsection 15.5.1).

Additional logic circuitry is added as shown in Figures 15D.2-6 and 15D.2-7. Also, an additional annunciator is added (Figure 15D.2-8) because the existing turbine trip alarm is produced by a limit switch on the turbine trip valve.

The total effect on the 238 Nuclear Island design is to improve safety. The operator is no longer required to manually reset the system following a high vessel water level trip to permit later operation if needed. He will no longer be distracted by the reset action and the possibility of inadvertent failure to reset is eliminated. The change

15D.2.1.4.1 Automatic RCIC Restart (Continued)

utilizes the steam supply valve to terminate steam flow on high water level only. The other five RCIC trip parameters will still close the turbine trip valve requiring manual reset of the system.

15D.2.1.4.2 RCIC Break Detection Logic Modification

This change is made in response to NUREG-0737 (Reference 1), Item II.K.3.15. The change increases the starting reliability of the RCIC system by reducing the likelihood of an inadvertent trip during system startup.

Existing System Operation

Each RCIC steam supply line is provided with two normally open isolation valves (E51-F063 and E51-F064). These valves close automatically upon receipt of an isolation signal. Each line contains a flow metering device located downstream of the isolation valves.

The flow sensing system will initiate closure of the isolation valves when the flow in that line exceeds 300% of rated. A pipe rupture can produce up to ten times rated flow. The issue raised by NUREG-0737, Item II.K.3.15 (Reference 1), is that the 300% setpoint may be momentarily exceeded during the RCIC start sequences causing unnecessary trip of the RCIC system and thus less than optimum reliability. Changing the setpoint would require extensive accident analyses involving the leak detection systems as well as the RCIC system. Addition of a time delay to the break detection circuitry directly addresses the problem and can be designed to have no impact on the currently documented accident analyses of RCIC steam supply line breaks (Subsection 15.6.4).

15D.2.1.4.2 RCIC Break Detection Logic Modification (Continued)

Hardware Changes

The design objectives are met by replacing the existing solid state isolation logic in each break detection circuit with time delay logic. A setpoint of at least 3 seconds, but less than the 13 seconds, will be utilized. This will involve no design changes in the differential pressure measuring devices. The RCIC system has two break detection circuits. Each circuit controls one of the two isolation valves. Both circuits in the system are to be modified. A discrete time delay circuit will be incorporated for the 238 Nuclear Island. (Figure 15D.2-9)

The timer is started when the flow rate sensed by the flow meters exceeds the trip setpoint. This setpoint is somewhat less than the analytical limit of 300% of rated flow. This difference provides margin for instrument errors and instrument drift and ensures that actual plant performance is within the scope of the assumptions used for the accident analyses. At the end of the timer period, system isolation will only occur if the flow meters are still reading at or above the trip setpoint. A variable 3 to 10 second time delay is planned. Preoperational testing will be performed to establish the setting for an individual plant (Section 14.2).

Effects of the Planned Change

The design objective of the RCIC isolation system is to limit the radiological consequences of a steam supply line rupture. The radiological consequences of such an accident are determined by the total quantity of fission products discharged to the environment (Section 15.4). Addition of a

15D.2.1.4.2 RCIC Break Detection Logic Modification (Continued)

time delay will not result in any change in the total reactor fluid mass release when the design basis conditions are considered. This is because a 13 second valve closure delay results from the assumption in design basis radiological calculations that no offsite AC power is immediately available. The diesel-generator start and emergency bus loading sequence is assumed to require 13 seconds and precludes any movement of the isolation valves prior to this time. The modification to the isolation system would still generate an isolation signal well before emergency power is available. There is thus no impact on the design basis analysis. Furthermore, unnecessary trips of the RCIC system will be avoided resulting in attendant improved reliability.

15D.2.1.4.3 ADS Logic Modification

The existing Automatic Depressurization System (ADS) actuation logic will be changed to respond to Item II.K.3.18 of NUREG-0737 (Reference 1). This change will automatically depressurize the reactor vessel for those events for which high pressure systems are unavailable or unable to maintain adequate water level, but do not result in a high drywell pressure trip (e.g., loss of feedwater with insufficient water delivery). Presently such an event requires manual initiation of ADS. By incorporating this change, the availability of low pressure water delivery systems is increased.

Existing Logic Design

The existing automatic depressurization system (ADS) logic design is shown in Figure 7.3-2. The design requires an initiation signal consisting of concurrent high drywell

15D.2.1.4.3 ADS Logic Modification (continued)

pressure and low reactor water level signals in order to actuate the ADS. The high drywell pressure signal is sealed into the initiation sequence and does not reset if the high drywell pressure subsequently clears. When both high drywell pressure and low water level signals have been received, the logic confirms the water level is indeed below the scram water level (to prevent spurious actuations) and starts the 120 second delay timer. The timer is automatically reset if the low water level trip clears before the timer times out; it can also be manually reset. The timer allows the operator time to bypass the automatic blowdown if reactor water level has been or is being restored, or if the signals are erroneous. To complete the sequence, the ADS logic receives a low pressure ECCS permissive based on pump discharge pressure to provide some assurance that makeup water will be delivered to the vessel once it is depressurized.

An event such as a loss of feedwater may not cause a high drywell pressure signal. The ADS system is manually initiated, if required, for such events.

Planned Logic Change

Details of this logic change are not finalized at this time. General Electric is currently reviewing several alternative logic changes which would accomplish the desired objectives. This evaluation includes an assessment of the reliability of each alternative in providing the initiation signal and avoidance of spurious initiation or other adverse effects. These results will be reviewed with the NRC, and once the final design change is selected by GE and approved by the NRC, it will be reflected in Subsection 7.3.1.1.1.2.]

15D.2.2 BWR Mitigation Features and Capabilities

This section describes the 238 Nuclear Island plant features which would mitigate the consequences of degraded core conditions in the extremely unlikely event of a severe accident. It is not expected that these features will be required to perform their functions since the systems described in Subsection 15D.2.1 are capable of preventing core damage. However, as further demonstration of the defense-in-depth approach taken in the 238 Nuclear Island design, the plant mitigation features and capabilities under postulated severe accident conditions are described below. These capabilities extend well beyond NRC requirements.

One of the most significant conclusions in the 238 Nuclear Island Probabilistic Risk Assessment (Section 15D.3) is in the area of accident mitigation. Specifically, the offsite consequences of a severe accident, even one postulated to involve loss of the primary containment integrity, are decades lower than previously estimated by WASH-1400 (Reference 3).

The 238 Nuclear Island containment employs a unique multi-building, multi-barrier design. The reactor vessel is enclosed in a steel and concrete drywell structure and surrounded by the pressure suppression pool. The drywell and suppression pool structures form the initial barrier around the reactor. These structures are fully enclosed in both a containment building and a shield building, which form a second and third barrier. (Figure 15D.2-10)

The function of the containment system in a nuclear power plant is to protect the public from excessive dose in the event of a severe accident. In the case of the 238 Nuclear Island, this function is accomplished in two ways. The

15D.2.2 BWR Mitigation Features and Capabilities (Continued)

first is through the multiple containment barriers, which are designed to maintain their integrity for all design basis events. Their design also provides sufficient margin to maintain their integrity for most events beyond the design basis. The second way of performing the containment function is through filtration of radioactive releases - which provides an additional level of protection for events well beyond the design basis.

Effective filtration, or scrubbing, of potential releases from the containment is an inherent safety feature of the Mark III pressure suppression containment. Filtration in a 238 Nuclear Island containment is provided by both the Standby Gas Treatment System and by the suppression pool. Potential releases from the primary system, resulting from degraded transient or accident events, pass through the suppression pool before reaching the containment building. The suppression pool effectively retains halogens and particulate fission products. The suppression pool retention is in addition to retention by natural plate-out mechanisms and containment sprays. These retention mechanisms are summarized in Figure 15D.2-11. The 238 Nuclear Island Probabilistic Risk Assessment has shown that the suppression pool scrubbing function will be maintained even in extreme accident sequences which might result in loss of integrity of the primary containment building. This arises from the substantial structural strength of the drywell and suppression pool. This provides high assurance that the containment function, protecting the public from excessive dose, would be performed even if the outer barriers (containment and shield buildings) were to lose their integrity.

15D.2.2 BWR Mitigation Features and Capabilities (Continued)

The mitigative capability of the 238 Nuclear Island Containment system for severe accidents can be quantified in terms of system pressure capability and fission product retention.

The pressure capability of various structures within the Mark III Standard Plant Containment System are listed in Table 15D.2-3 and are described in Section 15D.3, Appendix G. As shown in Table 15D.2-3, the ultimate pressure capability of the containment extends well above its design pressure. The structures with the highest pressure capability for dynamic and static overpressurizations are the drywell and suppression pool. Therefore, whatever the pressure challenge, suppression pool and drywell integrity would likely be maintained, thereby maintaining containment function. Fission product retention is provided by suppression pool scrubbing, and other natural retention mechanisms. Quantification of pool scrubbing factors is described in Subsections 15D.2.2.1.2 and 15D.2.2.3.

Figure 15D.2-12 shows the results of a realistic calculation of the offsite doses for a severe accident in which the particulate fission products have been effectively retained by the suppression pool. For this calculation a full core meltdown with no system recovery was assumed. Fission products were assumed to be released at a height of 40 meters four hours after event initiation. No credit was taken for evacuation of the population. The resulting doses are comparable to the 10CFR100 (25 rem) limit for design basis accidents. Therefore the realistic offsite doses for a severe accident in which all systems are assumed to fail and containment integrity is lost are comparable to doses

15D.2.2 BWR Mitigation Features and Capabilities (Continued)

conservatively calculated for design basis events where all safety systems continue to function with the exception of a single active component failure. Thus, the maintenance of containment function assures no adverse offsite health effects. Figure 15D.2-12 illustrates quantitatively how the mitigation features of the 238 Nuclear Island protect the public from excessive doses and reduce the risk from severe accidents.

15D.2.2.1 Effect of Suppression Pool Scrubbing on Severe Accident Consequences

The BWR/6 probabilistic risk assessment (Section 15D.3) demonstrates that fission products would be transported to the suppression pool for nearly all accident sequences. This section describes the effect of suppression pool scrubbing on the offsite consequences from severe accident sequences.

15D.2.2.1.1 PRA Results

The distribution of events which can contribute to the frequency of core damage are listed in Table 15D.2-4. Transient-initiated events contribute 99% of the assessed frequency of core damage of which 88% are initiated by loss of offsite power. Anticipated transients without scram (ATWS) contribute only 1.3 percent to the core damage frequency and the frequency of loss-of-coolant accidents (LOCA's) is negligible. Table 15D.2-5 lists the percent contribution to the assessed frequency of core damage by fission product release path and suppression pool condition. These results indicate that the most probable core damage event in a BWR/6 results in fission product releases through the safety

15D.2.2.1.1 PRA Results (Continued)

relief valves to a subcooled (condensing) suppression pool. These conditions would provide the highest amount of fission product scrubbing by the suppression pool. A very small fraction of the potential core damage events could result in discharge of the fission products into a saturated (non-condensing) suppression pool. Since a thermally saturated suppression pool represents a worst case condition for fission product scrubbing, GE's scrubbing tests were performed to simulate this condition. Even under these limiting conditions, the suppression pool was found to provide an extremely high fission product retention capability as discussed in Subsection 15D.2.2.3.

The next subsection summarizes the results of a survey of the available literature on suppression pool fission product scrubbing. The results of recent experiments performed by GE are provided in Subsection 15D.2.2.3 and Attachment A.

15D.2.2.1.2 Literature Survey of Suppression Pool Scrubbing Factors

15D.2.2.1.2.1 Introduction

In severe accident sequences, the presence of water in the fission product transport pathways provides an important means to minimize the quantity of airborne fission products. The 238 Nuclear Island uses the pressure suppression pool to provide a water barrier to fission product migration. Thus, significant retention of radioiodines and other fission products, except noble gases, is expected and must be accounted for in any realistic evaluation of accident consequences.

15D.2.2.1.2.2 Summary of Results of Literature Survey (Continued)

which the existing data base could support, and the potentially attainable DFs which could be supported by further testing, were presented for each dominant transport sequence in Reference 6.

15D.2.2.1.2.3 Conclusions from Literature Survey

Results of the literature survey (Reference 6) indicate that chemical forms similar to the inorganic iodides and particulates that would be expected to be released during postulated severe accidents would be retained in the suppression pool and would not escape into the primary containment air space.

Suppression pool decontamination factors that were found to be appropriate for use in BWR risk assessments, based on the literature survey, are presented in Table 15D.2-7 (repeated from Reference 6). Based on the data presented in NEDO-25420 and the expected BWR transport conditions, it was concluded that suppression pool decontamination factors of at least 10^2 for elemental iodine and 10^3 for particulate and cesium iodide could be expected for subcooled pools. For saturated pools, decontamination factors of at least 30 for elemental iodine and 10^2 for particulates and cesium iodide were justified by the literature data. NEDO-25420 also concluded that minimum values in Table 15D.2-7 would be increased several orders of magnitude by testing for the range of expected conditions during severe accidents. Testing has recently been completed by GE and the results, which are described in Attachment A, demonstrate that the suppression pool DFs are orders of magnitude higher than the lower bounds established by the literature survey.

15D.2.2.1.2.3 Conclusions from Literature Survey (Continued)

Other natural processes such as the agglomeration of particulates, plateout, deposition and washout also play an important role in limiting the quantity of fission products available for leakage to the environment. The overall attenuation factor applicable to BWR severe accident sequences includes both the effects of pool scrubbing and natural removal processes expected to occur in the various compartments of the 238 Nuclear Island.

15D.2.2.1.3 Effect of Pool Scrubbing Factors On Offsite Doses

A recent NRC review (Reference 5) of fission product releases in severe accidents concluded that all fission products other than noble gases and methyl iodide would be released as aerosols (i.e. particulates suspended in a gas phase of steam and noncondensables).

As concluded in Subsection 15D.2.2.1.1, postulated severe accidents leading to core damage in the 238 Nuclear Island result in the transport of fission products to the suppression pool. Fission product aerosols discharged to the pool must pass through 13 to 19 feet of water in the pool before reaching the primary containment airspace. Additionally, any aerosols escaping the pool would have to rise approximately 130 feet vertically to reach the most probable containment release path. Sequences which would allow all fission products to bypass the pool require failures in addition to the multiple failures which lead to core damage and are so highly unlikely as to be incredible. These bypass scenarios are discussed in Section 15D.3.

15D.2.2.2.1.1 Scrubbing Mechanisms (Continued)

(3) Brownian Diffusion (Continued)

$$k_d = 1.8\left[\frac{D_f}{VR^3}\right]^{1/2}$$

where:

k_d = diffusive absorption coefficient, cm^{-1}

D_f = diffusivity due to Brownian motion ($D_f = \frac{kTC_m}{3\pi\mu_g d}$), cm^2/sec

R = bubble radius, cm

k = Boltzmann constant, 1.38×10^{-16} $g\text{-}cm^2/sec^2$ °K

T = temperature in °K

The mobility of the particles decreases with increasing particle size, and the diffusive absorption coefficient is generally negligible compared to inertial and sedimentation adsorption coefficients for a particle size $\geq$1 μm.

15D.2.2.2.1.2 Overall Particle Absorption Coefficient and Decontamination Factor

The total theoretical particle absorption coefficient for the particle in a gas bubble scrubbing process is

$$K_i = k_{ni} + k_{si} + k_{di}$$

and the rate of particle absorption per unit path of the bubble in a water column is given by:

$$\frac{dC_i}{dl} = -K_i C_i$$

15D.2.2.2.1.2 Overall Particle Absorption Coefficient and Decontamination Factor (Continued)

Upon integration,

$$C_i = C_i^o \quad e^{-K_i L}$$]

where C_i = particle concentration of species i in gas bubbles at the outlet of a water column,

C_i^o = particle concentration of species i in gas bubbles at the inlet of a water column,

l = height of water column, cm

The decontamination factor (DF) for the scrubbing process is the ratio of the inlet concentration to the outlet concentration

$$(DF)_i = \frac{C_i^o}{C_i} = e^{K_i l}$$

For a mixture of particles with various particle sizes (but same particle density), the overall mass or activity DF can be calculated by:

$$(DF)_{overall} = \frac{1}{\Sigma_i (F_i/(DF)_i)}$$

where F_i = the mass or activity fraction c rticle size, i, in the mixture at gas b le inlet

15D.2.2.3 Suppression Pool Scrubbing Factors for Severe Accidents

15D.2.2.3.0 Introduction

Using the combined particulate scrubbing (Subsection 15D.2.2.2.1) and hydrodynamic model (Subsection 15D.2.2.2.2), it is possible to develop a method for calculating the decontamination factor expected in the suppression pool during severe accidents. A discussion of this methodology is presented here.

Hydrodynamic theory (Subsection 15D.2.2.2.2 and Attachment A) and tests demonstrated that the bubbles rise through the suppression pool in a swarm of small bubbles.

The decontamination factor for the bubble swarm is equal to the summation of the fractional contributions of all the bubbles. The decontamination factor for each bubble is governed by its scrubbing height. The simple calculational model which sums the decontamination factor contributions of the small bubbles in a swarm for a particle size, is shown in Figure 15D.2-14.

Three calculation cases are presented in Subsection 15D.2.2.3.2. The calculational procedure uses the equation given in Figure 15D.2-14 (where the terms have been previously defined in Subsection 15D.2.2.2.1) to calculate the decontamination factor as a function of particle size in the bubble swarm. The total decontamination factor (DF_T) is determined by summing the particle size DFs over the particle size distribution.

$$DF_T = \frac{1}{\Sigma_i (M_i/DF_i)}$$

15D.2.2.3.1 Model Inputs

Particle size distributions for corium-steel and corium-concrete experiments were obtained from Sandia Laboratories (References 12, 13) and were used for the calculations in Subsection 15D.2.2.3.2. They are shown in Figures 15D.2-15 and -16. Bubble volumes were calculated using the accident sequence dependent flow rates of steam and non-condensibles obtained from the MARCH Code (Section 15D.3, Appendix F).

15D.2.2.3.2 Calculated Scrubbing Factors For 238 Nuclear Island

Total decontamination factors using the particle size distributions in Figures 15D.2-15 and 16 and considering discharge into the suppression pool from x-quenchers and horizontal vents were calculated. The calculated results are shown in Table 15D.2-15.

The model calculation for the fission product scrubbing conditions is conservative (smaller DF than would be expected under actual accident conditions). Under the postulated accident conditions, steam and hydrogen would dominate the gas phase in the bubbles. The gas properties of a steam/hydrogen mixture were not taken into account in these model calculations. The absorption efficiencies due to particle inertia, sedimentation and diffusion should increase for steam/hydrogen compared to air. Furthermore, the effect of steam condensation is not included in the model. In a most recent study (Reference 14), the DF has been predicted to increase nearly linearly or exponentially, depending on the particle size, with increasing steam content in the gas bubbles.

15D.2.2.3.3 Conclusions Regarding Pool Scrubbing Factors

Using particle distributions from corium-steel and corium-concrete meltdown experiments, the estimated scrubbing factors are $DF>10^4$ for discharge through the quenchers and $DF\sim10^2-10^4$ for vent discharges. These results confirm that the BWR suppression pool would effectively retain fission product particles released under severe accident conditions.]

15D.2.2.4 Conclusions

Subsection 15D.2.2 provides a description of the 238 Nuclear Island features and capabilities which mitigate the consequences of postulated severe accidents. These capabilities extend well beyond NRC requirements.

The 238 Nuclear Island pressure suppression pool filters potential fission product releases, thereby limiting offsite doses and maintaining containment function.

The BWR/6 Standard Plant Probabilistic Risk Assessment has shown that suppression pool scrubbing will be maintained even in extreme accident sequences which might result in loss of integrity of the primary containment building. This results because of the substantial strength of the drywell and suppression pool structures.

Quantification of the fission product scrubbing capability of the suppression pool during severe accidents was accomplished by GE's Fission Product Scrubbing Program. This program resulted in the development of a first principles analytical model to describe pool scrubbing and the experimental verification of the model by mass-transfer and hydrodynamic testing. This model predicts that the suppression pool

15D.2.2.4 Conclusions (Continued)

would reduce particulate fission product releases by a factor of 10,000 in the unlikely event of a severe accident. These results confirm that the 238 Nuclear Island suppression pool would effectively retain fission products releases during severe accidents. This backup mitigative capability to maintenance of containment integrity provides further defense in-depth for assuring that the health and safety of the public is protected.

15D.2.3 Emergency Procedure Guidelines

One of the tasks undertaken by General Electric and the Boiling Water Reactor Owner's Group in response to post-TMI NRC requirements was to develop generic emergency procedure guidelines (EPGs) (Reference 2).

The first version of the EPGs (Revision 0) was developed for BWR/1-5 inventory threatening events and submitted to the NRC in June 1980. The EPGs were extended to cover BWR/6 in Revision 1 which was submitted to the NRC in January 1981. Revision 1 of the EPGs is applicable to the 238 Nuclear Island. Revision 2 in prepublication form was provided to the NRC on June 1, 1982. Revision 2 is also applicable to the 238 Nuclear Island.

The EPGs are symptom-based guidelines as opposed to event-based guidelines. The operator does not need to identify what event is occurring in the plant in order to decide on what actions to take. Rather, he observes the symptoms (utilizing a relatively few instruments) which are occurring and takes immediate actions based on responding to those symptoms. The symptom based EPGs provide a major

15D.2.3 Emergency Procedure Guidelines (continued)

simplification over previously developed procedures because for inventory threatening events, the operator needs only to maintain reactor vessel water level, independent of the reason for an initial water level decrease.

These symptom-based guidelines are a significant improvement since all of the existing emergency procedures used at BWRs, prior to development of the EPGs, were event dependent. If the operator did not have an emergency procedure for the event which was occurring at his plant, he may not have known the correct action to take. Operator confusion was possible since there were only a limited number of emergency procedures in place, but a large number of possible events. Symptomatic EPGs avoid that problem. There are a limited number of important symptoms and proper response is outlined in the EPGs. The specified operator actions depend upon the ability the operator has to monitor and respond to each symptom or instrument reading.

The other significant improvement to existing emergency procedures is that the EPGs have been developed with extensive GE - owner interaction. Realistic analysis has been used repeatedly when making decisions as to what steps are appropriate for inclusion in the EPGs. This analysis is used to decide on various action, levels for the symptoms and in identification of the success paths for total response to these symptoms.

The EPGs also go beyond the design basis of plant licensing. They are not limited to defined design basis events or single failures. The EPGs cover the range from all equipment (including non-safety grade systems) available to every injection system unavailable.

15D.2.3 Emergency Procedure Guidelines (continued)

A summary description of the content of NEDO-24934 (EPG Revision 1) follows.

EPG Revision 1 contains guidelines on level control, shutdown, and containment control.

The Level Control Guideline section provides guidance for operator use in restoring and stabilizing reactor pressure vessel (RPV) water level. The plant symptoms which alert the operator to enter this guideline are low RPV water level, high drywell pressure, or an isolation has occurred.

The shutdown section provides guidance for operator use in depressurizing the RPV to cold shutdown conditions. This guideline is entered from the Level Control Guideline after the RPV water level has been stabilized.

The Containment Control section provides guidance for the operator to use in controlling primary containment temperatures, pressure, and level whenever suppression pool temperature, drywell temperature, containment temperature, drywell pressure, or suppression pool water level are above their normal operating limit or suppression pool water level is below its normal operating limit. This guideline is executed concurrently with the guideline from which it is entered.

Cautions are noted at various points throughout the guidelines. These cautions clearly remind the operator of important instructions.

The EPG also contains contingencies for use when the symptoms are not being satisfactorily controlled as a result of actions taken from the main guidelines or as a result of

15D.2.3 Emergency Procedure Guidelines (continued)

equipment malfunctions. The contingencies included are Level Restoration, Rapid RPV Depressurization, Core Cooling without Injection, Core Cooling without Level Restoration, Alternate Shutdown Cooling, and RPV Flooding.

In summary, the EPGs are symptom-based guidelines which simplify operator response. They enable the operator to respond to symptoms rather than requiring that he diagnose the cause of the event. Their applicability extends beyond design basis events, and their development represents a significant improvement over past emergency procedures. The EPGs, in addition to the inherent 238 Nuclear Island features, provide an additional level of BWR operational safety relating to the prevention of severe accidents.

15D.2.4 References

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2. NEDO-24934, "Emergency Procedure Guidelines - BWR 1-6," General Electric Company, January 1981.

3. "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (October 1975).

4. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," June, 1974.

15D.2.4 References (Continued)

5. NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," USNRC, June 1981.

6. Rastler, D. M., "Suppression Pool Scrubbing Factors for Postulated BWR Accident Conditions", NEDO-25420, Class I, General Electric Company, June 1981.

7. Stratton, W. R. et.al, Letter to NRC Chairman John Ahearne on August 14, 1980.

8. Cohen, B. L. and Lee, I. S., "A Catalogue of Risks," Health Physics, 36 (1970) pp. 707-722.

9. Grosch, D. S. and Hopwood, L. E., "Biological Effects of Radiation", Academic Press, Inc. (1979).

10. N. A. Fuchs, "The Mechanics of Aerosols," translated from the Russian by R. E. Daisley and Morina Fuchs, Pergamon Press (1964).

11. S. Yuu, T. Jotake, and K. Abe, Power Technology, 17, 115-122 (1977).

12. NUREG/CR-0324/SAND 78-1511 "Light Water Reactor Safety Research Program, Quarterly Report, January-March, 1978," P. 17-24.

13. NUREG-0183-5/SAND78 - 0076, "Light Water Reactor Safety Research Program, Quarterly Report, July - September, 1977, P. 8-12.

15D.2.4 References (Continued)

14. D. C. Bugby, A. F. Mills, and R. L. Ritzman, "Fission Product Retention in Pressure Suppression Pools," Science Application, Inc., January 7, 1982.

15. BWR Emergency Procedure Guidelines (prepublication form) submitted in Letter BWROG 8219 dated 6/1/82 from T. J. Dente (BWR Owners Group) to D. G. Eisenhut (NRC).

TABLE 15D 2-1

BWR/6

WATER INJECTION SYSTEMS CAPABILITY

High Pressure	No. of Pumps	Total Flow Per Pump (gpm)	Necessary Conditions
HPCS*	1	1550	
RCIC	1	800	
FW	2	17000	Offsite Power & Condensate
CRD	2	85	Offsite Power
SLC	2	43	
Low Pressure			
HPCS*	1	6000	
RCIC	1	800	RPV Pressure > 50 psi
FW	2	15000	Offsite Power
CRD	2	85	Offsite Power
SLC	2	43	
LPCS*	1	6000	
LPCI*	3	7100	
FW Condensate	3	11500	Offsite Power
RHR Service Water	2	300	

*Emergency Core Cooling Systems (ECCS)

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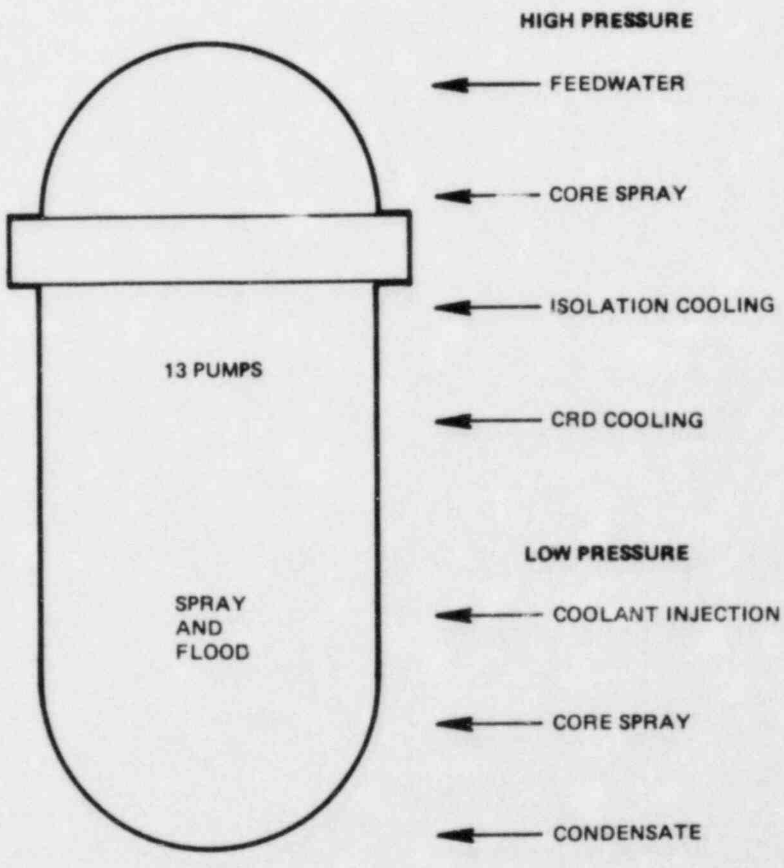
TABLE 15D.2-2 - MARK III LONG-TERM PRESSURIZATION

Time After LOCA sec (hr)	Total Btu Added to Suppression Pool 10^6 Btu	Suppression Pool Water Mass 10^6 lbm	Suppression Pool Temp °F	Containment Pressure psia
0.	0.	8.06	80.	14.7
200. (0.06)	847.*	11.29	141.	14.7
1731. (0.48)	952.	11.29	150.	15.0
12294. (3.4)	1361.	11.28	185.	15.6
25694. (7.1)	1756.	11.26	218.	16.3
98544. (27.4)	3349.	11.08	250.**	30.
171244. (47.6)	4591.	10.84	305.**	72.

*Includes:

Upper Pool Energy	=	196×10^6
All Vessel Inventory	=	334×10^6
120 sec. of Rated Feedwater	=	206×10^6
Vessel Sensible Heat	=	85×10^6
200 sec. of Decay Heat	=	26×10^6

**Assumed at Saturation temperature for containment pressure.



ACTUAL BWR CAPABILITY

Figure 15D.2-1. Systems to Supply Water to Core

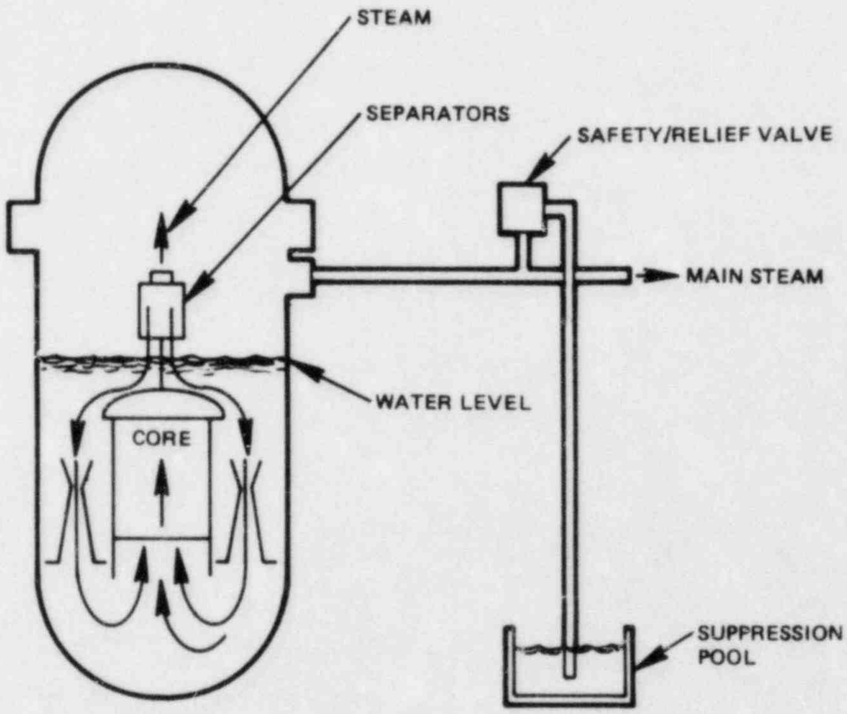


Figure 15D.2-2. Decay Heat Removal

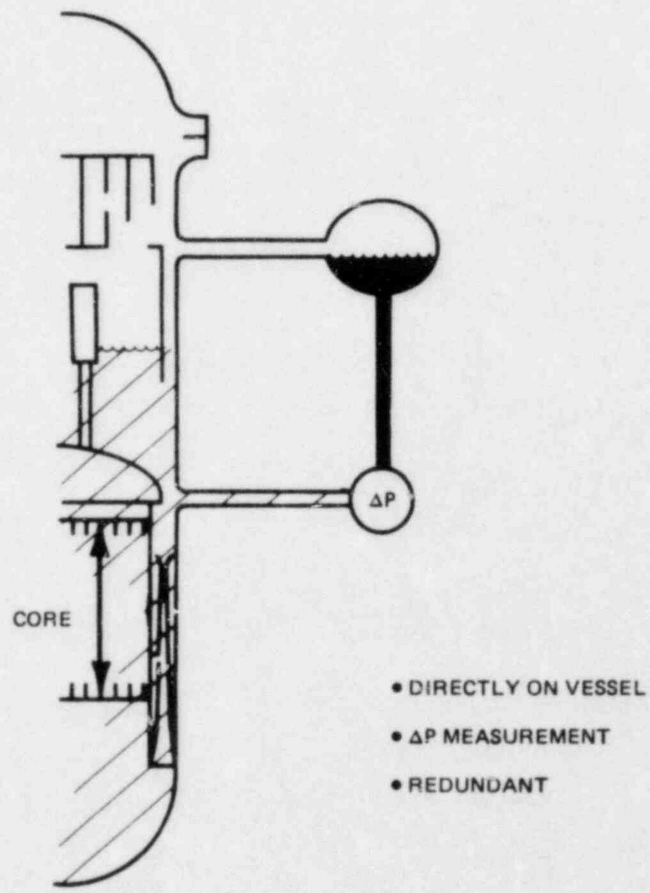


Figure 15D.2-3. Water Level Measurement

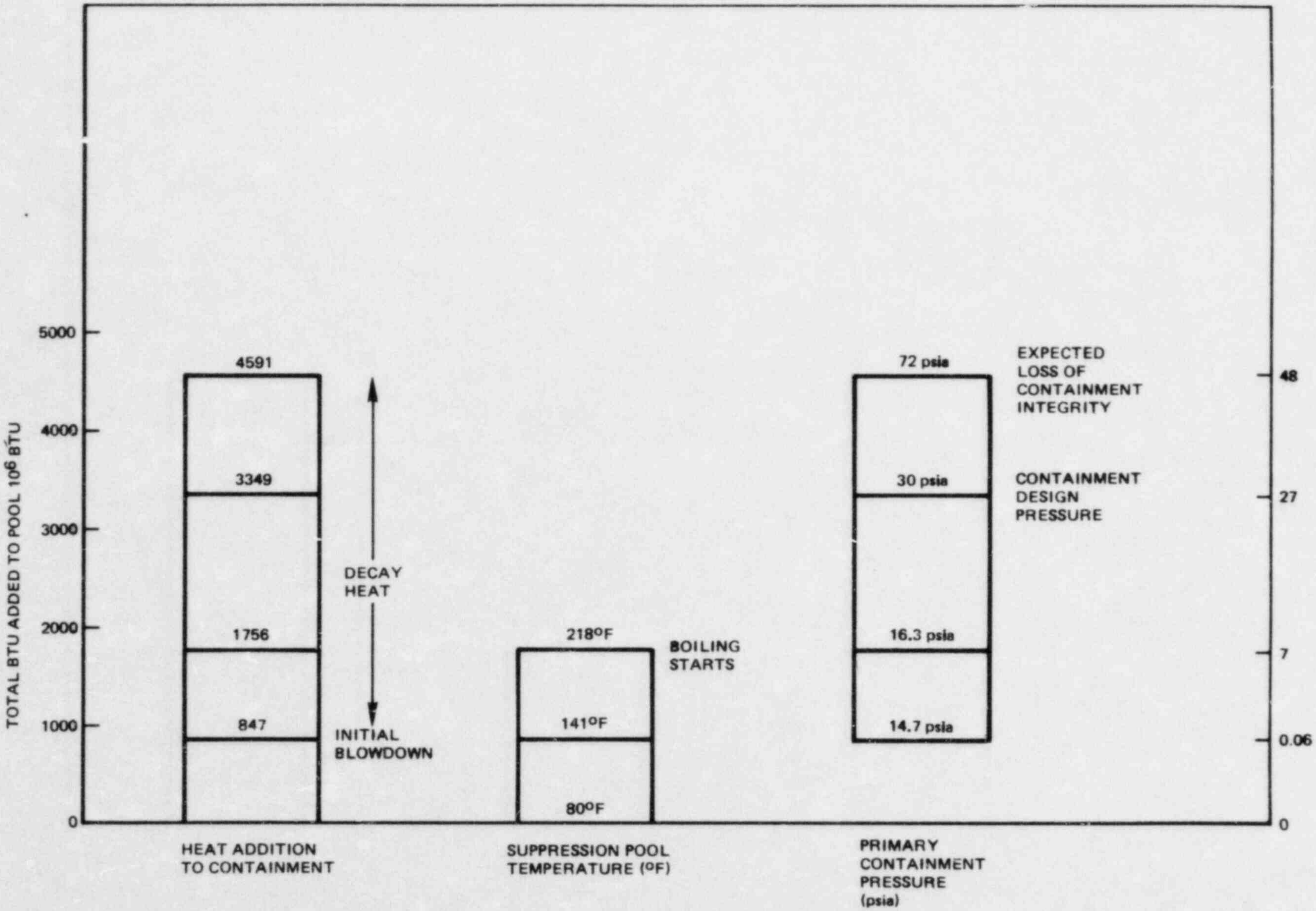
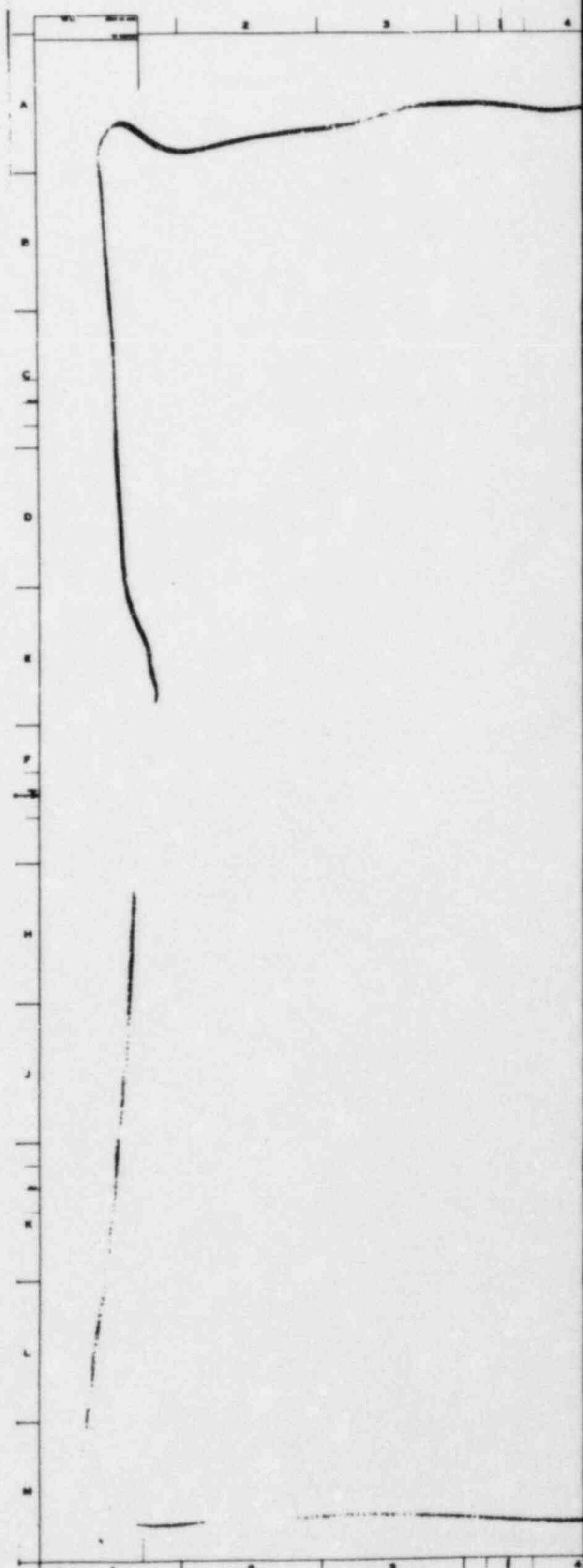


Figure 15D.2-4. Containment Passive Heat Sink Capability



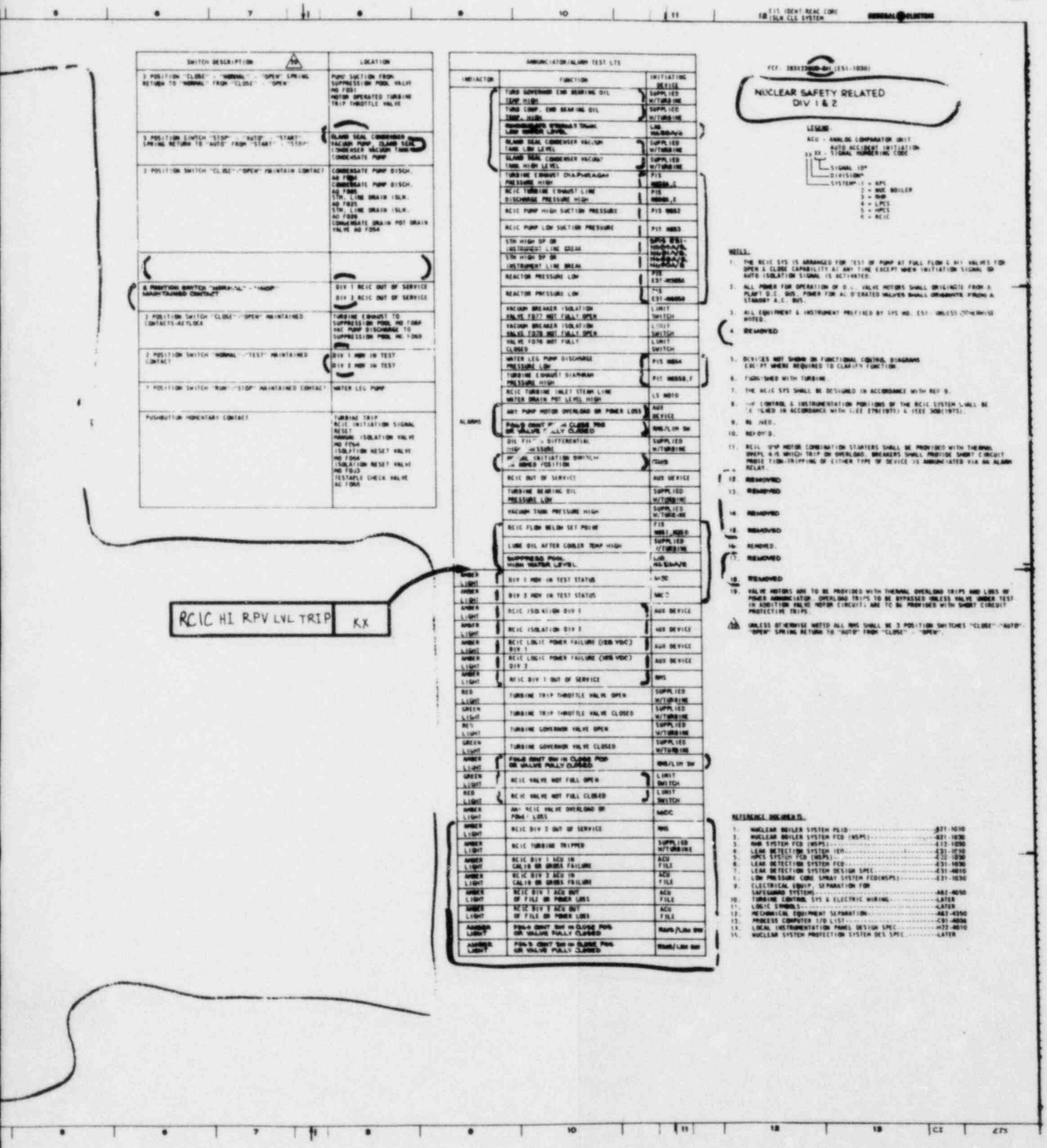
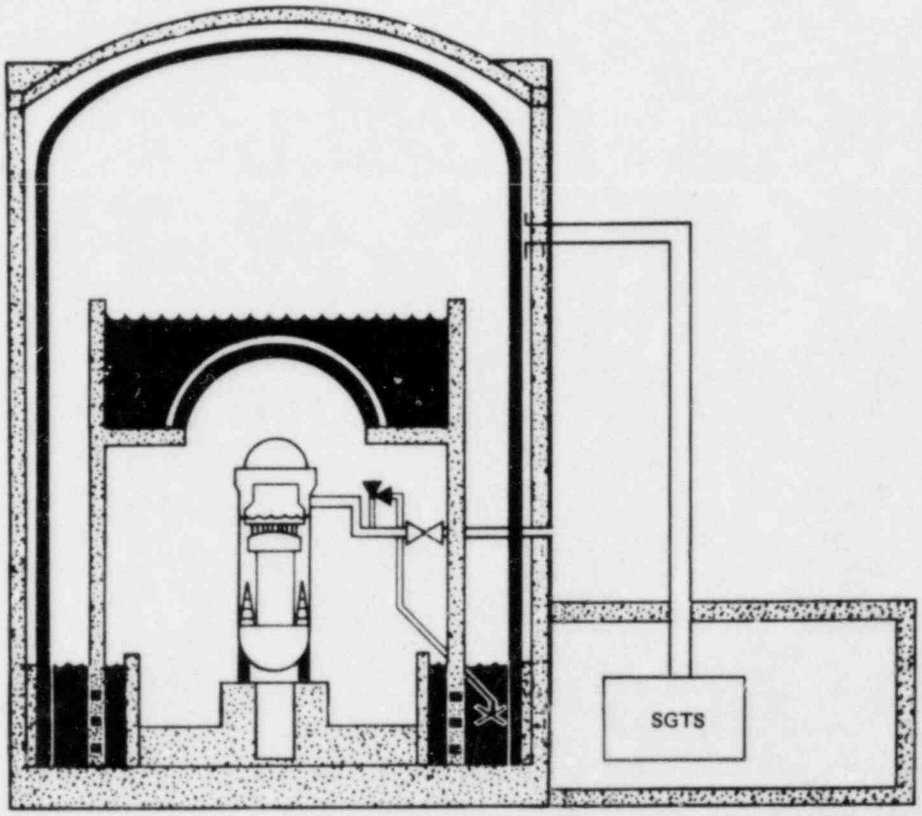


Figure 15D.2-8. Planned RCIC Restart Modification (Figure 7.4-1a)



	REGULATORY BASIS	ACTUAL CAPABILITY
DECONTAMINATION FACTORS		
• PLATEOUT	2	10
• SUPPRESSION POOL	1	100-10000
• CONTAINMENT SPRAYS	2	10-10000
• SECONDARY CONTAINMENT	5	5
• STANDBY GAS TREATMENT SYSTEM	100	1000
TOTAL	2000	50,000,000

]

Figure 15D.2-11. Mark III Fission Product Retention (Halogens and Particulates)

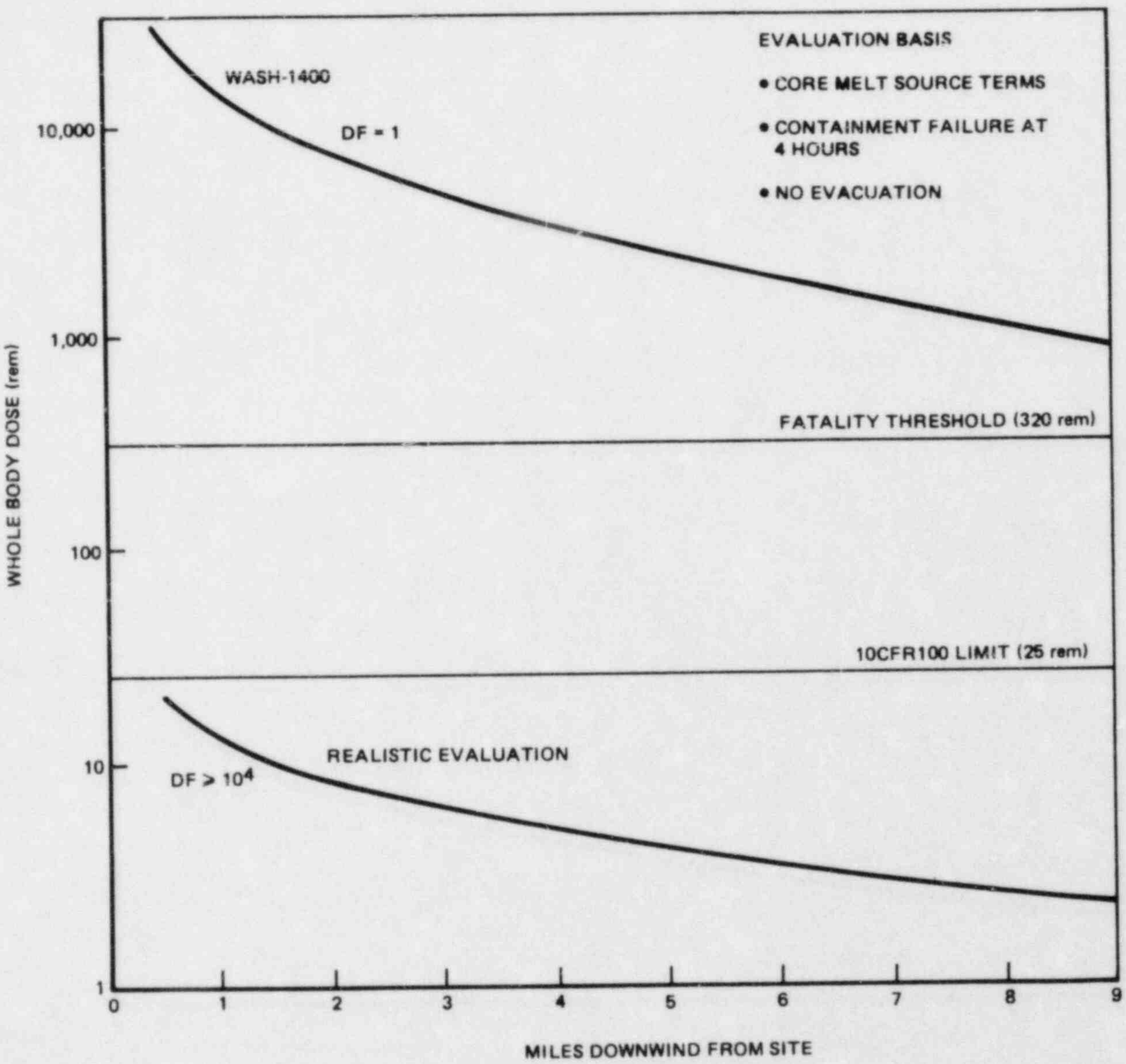


Figure 15D.2-12. Importance of Fission Product Retention in Mark III Pressure Suppression Containment

15D.4.2.1 Comparison of PRA Results to Draft Safety Goal (Continued)

Using the PRA results in Section 15D.3, comparison to the NRC proposed guidelines is provided in Table 15D.4-1. Comparison is made to all the numerical guidelines dealing with mortality risks and plant performance.

The calculated core melt probability of $\sim 5 \times 10^{-6}$ for the 238 Nuclear Island is a factor of 20 below the proposed guideline. As noted in Section 15D.3, there were no calculated early (prompt) fatalities, consequently, the 238 Nuclear Island design results are well below the NRC guidelines for individual and societal prompt fatality risks. The NRC numerical guideline for individual latent fatality risk is based on 0.1% of national statistics and is equivalent to $\sim 2.0 \times 10^{-6}$. The 238 Nuclear Island value is more than four orders of magnitude below this guideline. The PRA result of 2×10^{-4} latent fatalities (the mean of the risk curve for fatalities within 500 miles) when divided by the population within 50 miles of the site ($\sim$8.2 million people) yields the individual latent fatality risk of 2.5×10^{-11}. The societal latent fatality risk from Section 15D.3, 2×10^{-4} latent fatalities, is a factor of 10^{-4} below the guideline value of 3.2.

These results illustrate the capabilty of the 238 Nuclear Island design, as quantified in Section 15D.3, to meet the proposed NRC guidelines with extensive margin.

15D.4.2.2 Consideration of Hydrogen Control and Containment Design Changes

One of the key issues identified by the NRC related to severe accidents has been hydrogen control. In the notice of Proposed Rulemaking on Interim Requirements Related to Hydrogen Control in December, 1981, the NRC has proposed that additional hydrogen control systems be added to the BWR/6 - Mark III design to accommodate hydrogen release from postulated degraded core accidents. The proposed rule would require applicants to demonstrate maintenance of containment integrity for events which release an amount of hydrogen equivalent to 75% metal-water reaction of the active fuel cladding. GE has provided detailed responses to this proposed rule (Reference 3).

The Probabilistic Risk Assessment in Section 15D.3 assessed hydrogen generation for severe accidents. Further, the PRA quantified the consequences of hydrogen combustion events taking account of the structural capability of the drywell and pool to assure pool scrubbing of potential releases even if containment integrity were lost. This provision of fission product retention via suppression pool scrubbing means that containment function is maintained even for severe accidents. GE believes that maintenance of containment function should be the acceptance criteria for hydrogen control following severe accidents. Only a minimal risk reduction could be realized by eliminating hydrogen combustion accident sequences. The net effect of precluding hydrogen combustion for certain degraded core accident sequences would be to shift the loss of containment integrity from the time of hydrogen combustion to the time of containment overpressurization from non-condensibles generated by the core-concrete interaction.

15D.4.2.2 Consideration of Hydrogen Control and Containment Design Changes (Continued)

This delay in the time of fission product release reduces the 238 Nuclear Island risk by less than 30%. This small reduction is a result of additional time for fission product decay.

Therefore, in relative terms, the addition of a hydrogen control system provides only a minimal risk reduction. On an absolute basis, the 238 Nuclear Island risk is already low compared to the proposed NRC Safety Goal (Subsection 15D.4.2.1), and thus the provision of an additional hydrogen control system is inappropriate.

In addition to hydrogen control, the NRC has considered other changes affecting the Mark III containment design for future plants. Specifically, the NRC has considered the appropriateness of increasing the Service Level C capability for all containment designs to 45 psig. The NRC has also considered the appropriateness of including one or more dedicated containment penetrations in order not to preclude future installation of systems to prevent breach of containment, such as a filtered vented containment system. The following paragraphs provide a discussion of the significance of these proposed changes relative to the existing capability of the 238 Nuclear Island.

The 238 Nuclear Island mitigation features were presented in Subsection 15D.2.2 and the containment design capability was presented in Section 15D.3 Appendix G. These containment evaluations show that, in addition to the ultimate pressure capability of the primary containment significantly exceeding 45 psig, the Service Level C capability for the drywell and pool structures also significantly exceeds 45 psig. Based on the drywell and suppression pool strength, it has been

15D.4.2.2 Consideration of Hydrogen Control and Containment Design Changes (Continued)

concluded that the fission product retention function of the containment will be maintained. Maintenance of this function is accomplished even for postulated severe accidents with loss of integrity of the primary containment. Thus, fission product retention will be assured as a result of the structural capability of the drywell and pool. A further increase of the prmary containment building structural capability would not significantly reduce the risk due to severe accidents, since the non-condensible gases generated from degraded cores may still ultimately pressurize the containment above any increased capability value.

With respect to assuring that penetrations for a filtered vent could be provided, it has been shown in Subsection 15D.2.2 that the pressure suppression pool effectively filters fission product releases from postulated severe accidents. The pool and drywell thus provide the same mitigation capability as a filtered vented containment system, even for events that proposed filtered vented containment system designs cannot accommodate.

Therefore, it is concluded that changes to the pressure capability and the provision of a dedicated penetration are inappropriate for the 238 Nuclear Island containment.

15D.4.2.3 Consideration of Proposed NRC Policy Statement (SECY-82-1)

In the proposed policy statement included as Attachment A to SECY-82-1 (Reference 2), implementation guidelines are offered for new Construction Permit (CP) applicants. Although these guidelines are only in draft form, they provide useful criteria for measuring the capabilities of the 238 Nuclear

TABLE 15D.4-1

COMPARISON OF 238 NUCLEAR ISLAND PRA RESULTS
TO PROPOSED NRC SAFETY GOALS

Criteria	Proposed NRC Guideline	238 Nuclear Island Result
Core Melt Probability	1.0×10^{-4}	$\sim 5.0 \times 10^{-6}$
Individual Prompt Fatality Risk	5.0×10^{-7} (1)	$0^{(5)}$
Individual Latent Fatality Risk	2.0×10^{-6} (1)	2.5×10^{-11} (4)
Societal Prompt Fatality Risk	1×10^{-4} (2)	$0^{(5)}$
Societal Latent Fatality Risk	3.2 (3)	2×10^{-4}

NOTES:

(1) 0.1% of National Fatality Statistics]
(2) Assuming 1 mile average population of [illegible] people.
(3) Assuming 50 mile average population of 1.[illegible] million people
(4) Using theoretical 2×10^{-4} deaths spread over the PRA site 6 - 50 mile population of 8.2 million people
(5) No prompt fatalities were calculated for the 238 Nuclear Island PRA

TABLE 15DA.1-1

PARTICLE SIZE DISTRIBUTION ON IMPACTOR STAGE

MAIN STREAM IMPACTOR (INLET)

PLATE NO.	NO. OF HOLES	HOLE DIAM, CM	CUNNINGHAM SLIP	PARTICLE DIAM, CM
0	264	0.16130	1.05103	3.196E-04
1	264	0.11810	1.08266	1.973E-04
2	264	0.09140	1.12369	1.318E-04
3	264	0.07110	1.18515	8.807E-05
4	264	0.05330	1.29861	5.461E-05
5	264	0.03430	1.65624	2.496E-05
6	264	0.02540	2.22490	1.372E-05
7	156	0.02540	2.87085	9.288E-06

Particle Density = 7.800 gm/cc
Flow Rate = 472.000 cc/sec
Temperature = 293.00 deg K

TABLE 15DA.1-2

PARTICLE SIZE DISTRIBUTION ON IMPACTOR STAGE

AEROSOL IMPACTOR (EXIT)

PLATE NO.	NO. OF HOLES	HOLE DIAM, CM	CUNNINGHAM SLIP	PARTICLE DIAM, CM
0	264	0.16130	1.07759	2.285E-04
1	264	0.11810	1.12663	1.400E-04
2	264	0.09140	1.19126	9.272E-05
3	264	0.07110	1.29010	6.113E-05
4	264	0.05330	1.47902	3.706E-05
5	264	0.03430	2.14666	1.588E-05
6	264	0.02540	3.40894	8.029E-06
7	156	0.02540	4.95946	5.117E-06

Particle Density = 7.800 gm/cc
Flow Rate = 944.000 cc/sec
Temperature = 315.00 deg K

TABLE 15DA.1-3

COMPARISON OF TEST MATRIX AND SEVERE ACCIDENT CONDITIONS

PARAMETER	RANGE TESTED	RANGE EXPECTED FOR SEVERE ACCIDENTS
Bubble Size (cm):	0.4 - 1.4	0.5 - 0.6
Particle Concentration (g/m^3):	0.02 - 5.5	>5.5
Submergence Height (cm):	34 - 167	411 - 573
Gas and Water Temperature (°C):	20 and 60	49 - 649
Particle Size Distribution (μm):	0.05 to 10	.05 - 10

]

TABLE 15DA.1-4

SUMMARY OF SCRUBBING TEST RESULTS

Test Date	Bubble Diam,cm	Orifice/ Cap (1)	Bubble Rate B/Min	Particle Conc (g/m^3)	Weight Height cm	Overall D.F.
EU_2O_3 - Millipore						
9/16	0.47	70/none	183	0.18	34.3	108
9/29	0.63	70/0.6	145	0.17	34.3	333
9/30	0.63	70/0.6	145	0.44	34.3	214
10/1	0.60	70/0.6	277	0.02	34.3	119
10/8	0.74	130/0.8	254	0.53	34.3	189
10/27	0.85	180/0.8	312	0.48	167.7	1170
10/28	0.85	180/0.8	318	0.91	167.7	1415
10/30	0.45	180/0.2	248	0.87	167.7	1251
11/3	0.45	180/0.2	48	0.48	167.7	719
EU_2O_3 - Impactors						
12/1	0.86	180/0.7	260	0.76	167.7	896
12/2	0.86	180/0.7	260	5.5	167.7	1260
12/8	1.41	Special	124	0.30	167.7	534
12/9	1.35	Special	140	1.8	167.7	1260
12/10	0.78	180/0.7	248	4.95	76.2	910
12/11	0.88	240/0.7	276	4.34	167.7	4157
12/14	0.88*	240/0.7	272	1.38	167.7	2270
12/15	0.88	240/0.7	304	**N.D.	167.7	928

* 60°C Water/60°C Gas
** Not Determined

(1) Orifice size in microns, cap size in centimeters

TABLE 15DA.1-5

EUROPIUM OXIDE PARTICLE SIZE DISTRIBUTION

Mass Fraction	Average Particle Size* (μ)
.001	.1
.009	.15
.04	.33
.05	.67
.1	1.13
.1	1.73
.1	2.46
.6	> 2.46

]

* $\rho = 7.8$ gm/cm^3

15DA.3 HYDRODYNAMIC THEORY

This section describes the hydrodynamic theory which explains the observations in the hydrodynamic experiments.

15DA.3.1 Bubble Breakup Mechanisms

The mechanisms which contribute to bubble breakup identified in this analysis involve inviscid flattening, aerodynamic shredding, Taylor instability, and Helmholtz instability.

Inviscid flattening refers to the bubble initial distortion after it is free from its charging source and before a significant wake develops below it. Aerodynamic shredding involves a pressure reduction on the bubble equator due to higher liquid velocity as the bubble moves upward, and subsequent tearing apart as surface tension forces are overcome. Taylor instability refers to the amplitude growth of interface waves as a gas bubble top surface supports liquid above in a gravity field. Helmholtz instability involves amplitude growth of the interface when fluids of different velocity and density flow in parallel streams.

Inviscid flattening of a bubble is shown sequentially in Figure 15DA.3-1. The bubble is free from its point of charging and its pressure is the average of its surrounding liquid. The bottom surface penetrates upward toward the top, tending to flatten the bubble. This phenomenon is crudely explained by the fact that the bubble pressure is uniform throughout the contained gas, whereas the total pressure of the surrounding liquid is higher at the bottom due to the hydrostatic head. However, since the bottom liquid also must be at the bubble pressure, its velocity must be higher than that of the top liquid.

15DA.3.1 Bubble Breakup Mechanisms (Continued)

It is seen in Figure 15DA.3-1 that the bottom surface almost catches the top when it has risen about one initial radius. The rise velocity of the top surface is about $0.5\sqrt{gR}$, and that of the lower surface is about $1.38\sqrt{gR}$. For a bubble of 1.0 ft. radius, the upper and lower surfaces rise at about 2.8 and 7.8 ft/s.

When a free bubble begins to rise, vortex generation occurs from viscous shear in the surrounding liquid, and a wake is created which ultimately limits the rise velocity. Figure 15DA.3-1 is based on inviscid theory and applies prior to wake formation. It is expected that the initial lenticular distortion occurs before a significant wake effect can occur.

Consider a somewhat flattened bubble rising at steady velocity V_∞ through stationary liquid as shown in Figure 15DA.3-2.

An observer on the bubble would see flow coming toward him at velocity V_∞. A stagnation point would occur at the top of the bubble, and velocity of fluid at the outermost edge would be V. Potential flow theory shows that if the rising object were a cylinder, the ratio V/V_∞ is 2.0. The flattened bubble of Figure 15DA.3-2 resembles the cross section of an ellipse on the top half with major and minor axes 2a and 2b, for which

$$V/V_\infty = 1 + a/b \qquad (1)$$

Bubble internal pressure is approximately equal to the stagnation pressure P_O, and average fluid pressure $\bar{P}$ on any quadrant roughly corresponds to the average velocity on a cylinder, namely V/π, for which

15DA.3.2 Bubble Breakup Distance (Continued)

than inviscid flattening alone. Bubble rise height with all breakup mechanisms active can be estimated from energy methods.

Suppose that a bubble of radius R_o and volume "V_o" is initially submerged below the pool surface a distance H, as shown in Figure 15DA.3-3. This configuration corresponds to a gas-liquid system with a given value of initial energy. As the bubble rises, the system energy is redistributed between fluid kinetic and potential energies, the energy associated with surface tension as new bubbles are formed, and liquid internal energy increase due to viscous drag effects.

If gas compressibility is neglected, the liquid potential energy associated with a spherical bubble submerged to depth H as shown in Figure 15DA.3-3 corresponds to the work of submergence,

$$E_i = \gamma V_o H \tag{11}$$

If the initial bubble has divided into n equal bubbles of radius r by the time it rises to elevation y , the sum of volumes is equal to the initial volume, which leads to

$$n = (R_o/r)^3 \tag{12}$$

The liquid potential energy for n bubbles at elevation y is given by

$$PE = n\ (4\pi r^3/3)\ \gamma\ (H-y) = V_o\ \gamma\ (H-y) \tag{13}$$]

the liquid bulk kinetic energy of a single bubble of radius r moving at velocity V is $(2\pi r^3/3)\gamma\ V^2/2g$. Therefore, the kinetic energy of n bubbles is

15DA.3.2 <u>Bubble Breakup Distance</u> (Continued)

$$KE = n\ (2\pi r^3/3)\ \gamma\ V^2/2g = (2\pi R_o{}^3/3)\ V^2/2g \qquad (14)$$

The initial bubble surface area is $4\pi R_o{}^2$. The increased area when n bubbles of radius r have formed is $n(4\pi r^2)$ so that the surface tension energy stored in the newly created surface area is

$$E_\sigma = \sigma\ (n4\pi r^2 - 4\pi R_o{}^2) = 4\pi R_o{}^2\ \sigma(R_o/r - 1) \qquad (15)$$

The increase of dissipation energy forms associated with vorticity and internal energy in the liquid is equal to that energy transferred as the rising bubbles perform viscous or drag work. The drag force of one bubble of radius r is given by $C_d \gamma \pi r^2 V^2/2g$. Therefore, as the number of bubbles increases, the dissipated energy is

$$E_\tau = \int_o^y (nC_d \gamma \pi r^2 V^2/2g)dy = C_d \gamma \pi R_o^3 V^2/2g \int_o^y \frac{1}{r}\, dy \qquad (16)$$

Since the total system energy must remain constant,

$$E_i = PE + KE + E_\sigma + E_\tau \qquad (17)$$

Assuming a constant average bubble velocity $dy/dt = V$, the derivative of E_i is written as

$$dE_i/dt = 0 = -V_o \gamma V + 0 - (4\pi\sigma R_o{}^3/r^2)dr/dt + C_d \gamma \pi R_o{}^3 V^3/2gr$$

15DA.3.2 Bubble Breakup Distance (Continued)

or,

$$dr/dt - (C_d \gamma V^3/8\sigma g)r = -(\gamma V/3\sigma)r^2 \qquad (18)$$

with the initial condition,

$$t = 0 \ , \ r = R_o \qquad (19)$$

and bubble elevation is given by

$$y = Vt \qquad (20)$$

A solution of Equation (18) combined with (19) and (20) yields

$$\frac{r}{R_o} = \frac{(3C_d V^2/8gR_o)}{1 + (3C_d V^2/8gR_o - 1)\exp(-(C_d \gamma V^2/8\sigma g)y)} \qquad (21)$$

Equation (21) gives the size of bubbles formed at elevation y .
As y increases, the bubble size becomes

$$r \to 3C_d V^2/8g \qquad (22)$$

If the rise velocity is 1 fps and the drag coefficient is between 0.5 and 1.0, corresponding to a bubble shape somewhere between a sphere and a disk, the average size of broken up bubbles would be about 0.24 inches in diameter. These would be formed, according to Equation (21), a distance of less than one inch after the wake forms and the initial bubble

15DA.3.2 Bubble Breakup Distance (Continued)

reaches a corresponding terminal velocity. If the initial bubble rises between one or two radii before a wake forms, one expects sudden division into many small bubbles immediately after that. The hydrodynamic tests show a large bubble breaks away from its charging source, after which the lower surface appears to snap through to the top, shattering the entire bubble.

This model simplifies the actual process by neglecting bubble interaction and incorporating the idealization of spherical bubbles with constant velocity and drag coefficients. However, it shows that even with energy dissipated by drag forces, and kinetic energy increase of the surrounding liquid, there is sufficient excess energy transfer to shatter the bubbles quickly.

15DA.3.3 Summary and Conclusions

This analysis examined breakup mechanisms of gas bubbles rising through liquid.

Free bubbles, which are at the average hydrostatic pressure, undergo breakup as they rise through liquid by buoyancy. Four bubble breakup mechanisms identified were:

(1) inviscid flattening during which the lower surface overtakes the upper rising surface;

(2) aerodynamic shredding, which pulls the bubble apart in a horizontal plane by higher liquid velocity past the bubble equator, correspondingly lower pressure, and a resulting outward force which overcomes surface tension;