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NRC-94-0073

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

References: 1) Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

2) NRC Letter to Detroit Edison, "Fermi 2 - Transmittal of Preliminary Accident Sequence Precursor (ASP) Analysis of Event for Licensee Peer Review," dated July 6, 1994

Subject: Response to Request for Review of ASP Analysis of Fermi 2 Event

Enclosed is the requested review (Reference 2) of the Accident Sequence Precursor (ASP) analysis of the Fermi 2 December 25, 1993 turbine event. Note that the review produced several recommendations for modifications to the ASP analysis.

If you have any questions concerning the review, please contact Mr. Earl M. Page at (313) 586-4266.

Sincerely,

Enclosure

cc: T. G. Colburn
M. P. Phillips
K. R. Riemer
NRC Regional Administrator

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Review of the Preliminary Accident Sequence Precursor (ASP) Analysis of the Fermi 2 December 25, 1993 Turbine Event Described in LERs 341/93-014 and 015

I. Introduction

A review was made of the subject ASP analysis. The review focused on the event description and modeling assumptions as they related to the specific event. No attempt was made to critique the details of the generic ASP identification or quantification approach as given in Appendix A to NUREG/CR-4674, Vol. 17. As a result of this review, several recommendations for modifications to the analysis are described below in Section II.

It should be noted that a preliminary analysis of this same event was completed inhouse in early 1994 using the Fermi 2 Individual Plant Examination (IPE) PSA model. This activity is briefly discussed in Section III.

Also, note that the ASP package as received contained two copies of LER 341/93-014-01 rather than one each of LER 341/93-014-01 and 015-01 as stipulated in the enclosure listing.

II. Recommended Revision to Preliminary ASP Analysis

The following changes are recommended for the modeling approach used for the December 25, 1993, Fermi 2 Turbine Event as described in the preliminary ASP report for LERs 341/93-014 and 015.

1. LPCI should be considered operable.

   Rationale:

   - Statements made concerning the inoperability of Low Pressure Coolant Injection (LPCI) due to the failure of valve B3105-F031B to close are largely based on the Technical Specifications approach

to operability. Full design flow in the event of a design basis Loss of Coolant Accident (LOCA) would not have been available without corrective action. However, for the non-LOCA transient event in question, including failure of F031B, LPCI would have injected full design flow into the vessel (following depressurization) had it been needed to restore inventory lost to boiloff from decay heat. Note that at least 15 percent of design LPCI flow would be available even in the case of a LOCA and valve F031B stuck open. (See LER 93-015-01.)

2. The standby feedwater (SBFW) system needs to be identified as a high pressure injection option. To be consistent with the Fermi 2 IPE treatment of high pressure injection capability, the SBFW function would replace the Control Rod Drive (CRD) hydraulic system top in the event tree of Figure 2. (However, note that CRD flow was credited for low pressure conditions in the IPE.)

   Rationale:

   - The Fermi 2 plant includes a manually initiated, non-safety related Standby Feedwater System (SBFW) employing two motor driven pumps, each capable of delivering about 600 gpm at high pressure through the normal feedwater inlet. The SBFW system is described in the UFSAR (Section 10.4.8, attached). Its purpose is to provide additional assurance of the capability to maintain reactor core cooling. However, it is not required in the licensing basis to support the safe shutdown of the reactor except for its use in the alternate shutdown system to meet Appendix R requirements. Its use is also specified in the Emergency Operating Procedures (excerpts from "Reactor Water Level Control", EOP 29.000.01, attached) and it was included in the Fermi 2 IPE. Moreover, it was actually used in the subject turbine event for water level control as described in LER 93-014-01.

3. Rather than considering one train of shutdown cooling (SDC) completely inoperable due to failure of valve F031B, it would seem more appropriate to apply a recovery factor to the assumed inoperable train as done for LPCI injection.

   Rationale:

   - The above recovery recommendation was actually employed the day following the event. It is recognized that if SDC were required

due to unavailability of other heat removal paths, it would have been required sooner than actually used. However, for a decay heat containment heat-up transient, there should still be ample time for a high chance for successful recovery (alternate valve lineup).

4. Include the hard pipe vent as an alternate heat removal option.

Rationale:

- As with all BWR4's, there is hard-piped containment venting capability that can be used as a last resort for heat removal. Its use is stipulated in the EOPs (excerpts from Primary Containment Pressure Control, EOP 29.000.02, attached), it is described in the UFSAR (Section 6.2.5.2.5.1, attached), and it is included in the Fermi 2 IPE. All necessary support systems for vent operation were available during the turbine event.

III. In-House Analysis of the Fermi 2 Turbine Event

A preliminary determination of the conditional core damage frequency for the subject turbine event was performed by making appropriate modifications to the Level 1 PSA model used for the Fermi 2 IPE. A single initiating event, "turbine trip without condenser" was run with the appropriate additional failures. The resulting conditional CDF was 2.6×10^{-5}.

An additional case was run to treat the potential for failure of the Standby Feedwater System (SBFW) which is located in the basement of the Turbine Building where extensive flooding occurred. While the SBFW did not fail, one might argue it was at risk and variations in flooding behavior could have rendered it inoperable. To bound this indeterminate risk, an additional case was run that simply assumed the SBFW system to be inoperable. The resulting conditional CDF was 5.9×10^{-5}.

ATTACHMENTS

In the feedwater heating portion of the system, temperature measurements are provided for each stage of heating. Steam pressure measurements are provided at each feedwater heater.

Instrumentation and controls are provided for regulating the heater drain flow rate in order to maintain the proper condensate level in each feedwater heater shell or heater drain tank. High-level alarm and automatic emergency drain action on high level are also provided.

10.4.8 Standby Feedwater System

10.4.8.1 Design Basis

The standby feedwater (SBFW) system provides condensate from the condensate storage tank to the feedwater system downstream of the No. 6 feedwater heater. It is a manually initiated system to provide additional assurance of the capability to maintain reactor core cooling and to prevent the uncovering of the core. No credit for the SBFW system has been assumed in the accident analyses in Chapters 6 or 15. The system may be initiated by the control room operator in response to an operational transient, e.g., loss of normal feedwater. This minimizes demands on other high-pressure core cooling systems. The system is not safety related and is nonseismic.

10.4.8.2 System Description

The SBFW system consists of piping, valves, pumps, motors, controls, instrumentation, and associated equipment that supply the feedwater system with condensate from the condensate storage tank. There are two SBFW pumps with a nominal capacity of 1300 gpm and 1247 psig. Each pump is driven by a 700-hp motor; the motors are independently fed from the SS64 and SS65 transformers. The pumps discharge to two parallel motor-operated (dc) modulating flow control valves. The larger valve (6 in.) is used when reactor pressure is near 1120 psi; the other (4-in.) valve is used when reactor pressure is low. There is a motor-operated (dc) isolation valve before tying into the feedwater system. This valve will automatically open when either pump is started and will close at RPV Level 8. The system diagram is shown in Figure 10.4-11.

10.4.8.3 Safety Evaluation

The SBFW system is not required to support the safe shutdown of the reactor except for its use in the alternate shutdown system to meet 10 CFR 50, Appendix R, Section III.L. See Subsection 7.5.2.5. (Inadvertent initiation of the system is bounded by the inadvertent high-pressure-coolant-injection (HPCI) transient, discussed in Subsection 15.5.1, since HPCI flow is approximately five times SBFW flow.)

10.4.8.4 Tests and Inspections

Normal manufacturer's tests were performed on the SBFW pumps and motors. Prior to initial operation, the system received a field hydrostatic test and inspection in accordance with ANSI N18.7.6.

10.4.8.5 Instrumentation Application

Controls are located in the main control room. Measurement of pump discharge flow is provided in the main control room. Pump, motor bearing, and winding temperatures are recorded and alarmed by the main control room data logger.

RPV CONTROL

Revision Summary:

1) Added information on actuations and isolations to the Automatic Actuations and Isolations Table in Supplemental Information.
2) Incorporated TCN T08356 which revised ARI Logic Defeats.

Implementation Plan:

1) This procedure goes into effect upon approval.
2) A summary of this revision will be placed in an Operations Required Reading Package.
3) No further training is required.

29.000.01

RPV CONTROL (RC)

PURPOSE

The purpose of this EOP is to:

- o Maintain adequate core cooling,
- o Shut down the reactor, **and**
- o Cool down the RPV to cold shutdown conditions (RPV water temperature between 80°F and 200°F).

ENTRY CONDITIONS

The entry conditions for this EOP are **any** of the following:

- o RPV water level **below** 173 in.
- o RPV pressure **above** 1093 psig
- o Drywell pressure **above** 1.68 psig
- o A condition which requires reactor scram **AND** reactor power **cannot** be determined to **be below** 3%.

OPERATOR ACTIONS

RC-1 **IF** reactor scram has **not** been initiated,
THEN initiate reactor scram.

RC-2 Execute the following sections **concurrently**, **irrespective** of the entry conditions:

- o RC/L, page 5
- o RC/P, page 27
- o RC/Q, page 41

RPV WATER LEVEL CONTROL (RC/L)

RPV WATER LEVEL CONTROL (RC/L)

================================ CAUTION ================================
RPV water level indication may be unreliable while executing section RC/L (RC Enclosure, page 76).
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RC/L-2 Restore **and** maintain RPV water level between 173 in. and 214 in. with one or more of the following systems:

- o Condensate/feedwater, taking Bypass Valve Mode Switch to START if necessary (23.107)
- o Standby Feedwater (23.107.01)
- o CRD, operating both pumps if necessary (RC Enclosure, page 123)

======================== CAUTION ========================
Operating HPCI or RCIC turbines below 2100 RPM may result in unstable system operation and equipment damage.

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Elevated torus pressure may trip the RCIC turbine on high exhaust pressure.
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- o RCIC with suction from the condensate storage tank (23.206), defeating low RPV pressure isolation interlocks if necessary (RC Enclosure, Section 1, page 133).
- o HPCI with suction from the condensate storage tank (23.202), defeating high torus water level suction transfer logic if necessary (RC Enclosure, Section 3, page 137).

(continued)

PRIMARY CONTAINMENT CONTROL (PC)

Revision Summary

1) Added information on actuations and isolations in the Automatic Actuations and Isolations Table in Supplemental Information.

Implementation Plan

1) This revision is effective upon approval.
2) A summary of these changes will be placed in Operations Required Reading.
3) No further training is required.

ARMS - INFORMATION SERVICES

Date approved: 12-08-93 Release authorized by: Daisy Shelton

Change numbers incorporated: 93-1656

DSN 29.000.02 Rev 20 Date 12-10-93

DTC TPNPP File 1703.02 Recipient 696

29.000.02

PRIMARY CONTAINMENT CONTROL (PC)

PURPOSE

The purpose of this EOP is to:

- o Maintain primary containment integrity, and
- o Protect equipment in the primary containment.

ENTRY CONDITIONS

The entry conditions for this EOP are **any** of the following:

- o Torus water average temperature **above** 95^{O}F
- o Drywell average temperature **above** 145^{O}F
- o Drywell pressure **above** 1.68 psig
- o Torus water level **above** +2 in.
- o Torus water level **below** -2 in.
- o Primary containment hydrogen concentration **above** 1%

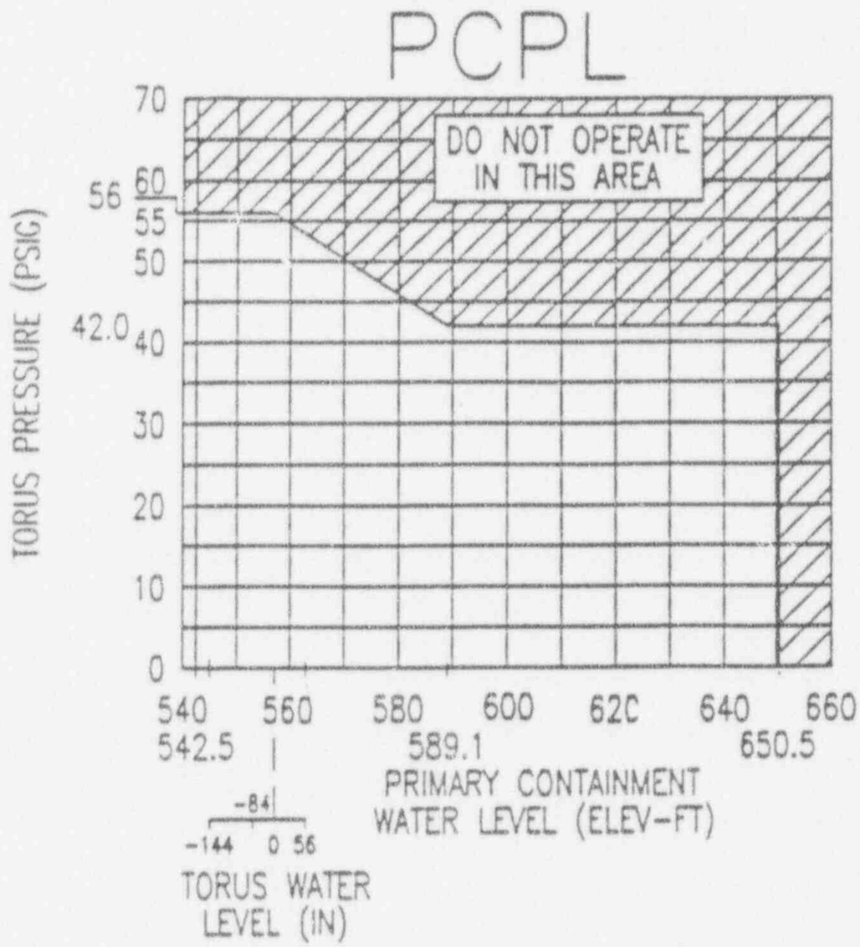
OPERATOR ACTIONS

PC-1 Execute the following sections **concurrently, irrespective** of the entry condition:

- o TW/T page 5
- o DW/T page 11
- o PC/P page 15
- o TW/L page 25
- o PC/H page 33

PRIMARY CONTAINMENT PRESSURE CONTROL (PC/P)

PC/P-5 BEFORE torus pressure reaches the Primary Containment Pressure Limit,
THEN vent the primary containment **irrespective** of offsite radioactivity release rate, defeating interlocks if necessary, to reduce and maintain pressure **below** the Primary Containment Pressure Limit, **as** follows:



- IF torus water level is below an elevation of 573 ft (PC Enclosure, page 133),
  THEN vent the torus in accordance with Primary Containment Venting and Purge Procedure (PC Enclosure, Section 3, page 87).

- IF torus water level is **at or above** an elevation of 573 ft (PC Enclosure, page 133),
  OR
  the torus **cannot** be vented,
  THEN vent the drywell in accordance with Primary Containment Venting and Purge Procedure (PC Enclosure, Section 3, page 87).

EMERGENCY PRIMARY CONTAINMENT VENTING

SECTION 3 Emergency Primary Containment Venting (to Reduce Pressure when Approaching the Primary Containment Pressure Limit Irrespective of Offsite Release Rates)

1.0 **WHEN** directed to enter this section by PC/P-5
THEN continue in this section

NOTE: All controls and indications are on COP H11-P808 or COP H11-P817 unless noted.

2.0 Initiate or verify initiation of SGTS (Section 1, Page 79). ___

==========================CAUTION==============================

Venting the Primary Containment may release radioactive gas/steam into the Reactor Building.

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One division of SGTS must be shutdown while venting to ensure at least one SGTS train is operable if a LOCA occurs.

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3.0 Verify shutdown or shutdown one division of SGTS. ___

4.0 Contact Radiation Protection to sample the primary containment atmosphere. ___

5.0 **IF** torus is directed to be vented
THEN vent the torus in accordance with step 7. ___

6.0 **IF** drywell is directed to be vented
THEN vent the drywell in accordance with step 8. ___

7.0 Vent from the torus defeating interlocks if necessary (Section 6, page 125). ___

7.1 Open 1 inch pressure control valves
- o T48-F457, Pressure Control Inlet Isolation Valve
- o T48-F458, Pressure Control Exhaust Isolation Valve

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7.2 **IF** primary containment pressure is reduced to 40 to 45 psig,
THEN close:
- o T48-F458, Pressure Control Exhaust Isolation Valve
- o T48-F457, Pressure Control Inlet Isolation Valve

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7.3 **IF** primary containment pressure cannot be reduced,
THEN open the 2 inch vent path:
- o T48-F459, Bypass Leakage Isolation Valve
- o T48-F409, Suppression Chamber N_2 Supply Isolation Valve
- o T48-F404, Suppression Chamber Inlet Isolation Valve

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Intentionally
Blank

EMERGENCY PRIMARY CONTAINMENT VENTING

7.4 **IF** primary containment pressure is reduced to 40 to 45 psig,
THEN close:

- o T48-F404, Suppression Chamber Inlet Isolation Valve
- o T48-F409, Suppression Chamber N_2 Supply Isolation Valve
- o T48-F459, Bypass Leakage Isolation Valve
- o T48-F458, Pressure Control Exhaust Isolation Valve
- o T48-F457, Pressure Control Inlet Isolation Valve

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7.5 **IF** primary containment pressure cannot be reduced,
THEN shutdown SGTS **and** close or verify closed the following SGTS dampers/valves:

- o T46-F008A, Secondary Containment Outboard Isolation Damper
- o T46-F409, Secondary Containment Inboard Isolation Damper
- o T46-F008B, Secondary Containment Outboard Isolation Damper
- o T46-F408, Secondary Containment Inboard Isolation Damper
- o T46-F407, Reactor Building Exhaust System Isolation Valve
- o T46-F406, HPCI Vacuum Pump Exhaust Isolation Valve
- o T46-F410, Air Inlet Refueling Area Isolation Valve

and

open or verify open:

1. T46-F420, Hardened Vent Inboard Isolation Valve
2. T46-F421, Hardened Vent Outboard Isolation Valve
3. T46-F412, SGTS Suppression Chamber Purge Line Isolation Bypass Valve (6 inch line)
4. T46-F400, Suppression Chamber Purge Isolation Valve

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7.6 **IF** primary containment pressure is reduced to 40 to 45 psig,
THEN close:

1. T46-F400, Suppression Chamber Purge Isolation Valve

- o T46-F412, SGTS Suppression Chamber Purge Line Isolation Bypass Valve
- o T48-F457, Pressure Control Inlet Isolation Valve
- o T48-F458, Pressure Control Exhaust Isolation Valve
- o T48-F404, Suppression Chamber Inlet Isolation Valve
- o T48-F409, Suppression Chamber N_2 Supply Isolation Valve
- o T48-F459, Bypass Leakage Isolation Valve
- o T46-F420, Hardened Vent Inboard Isolation Valve
- o T46-F421, Hardened Vent Outboard Isolation Valve

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EMERGENCY PRIMARY CONTAINMENT VENTING

7.7 **IF** primary containment pressure cannot be reduced,
THEN open T46-F401, SGTS Suppression Chamber Purge Line Isolation Valve (20 inch line)

7.8 **IF** primary containment pressure is reduced to 40 to 45 psig
THEN close or verify closed:

1. T46-F400, Suppression Chamber Purge Isolation Valve
- o T46-F401, SGTS Suppression Chamber Purge Line Isolation Valve
- o T46-F412, SGTS Suppression Chamber Purge Line Isolation Bypass Valve
- o T48-F404, Suppression Chamber Inlet Isolation Valve
- o T48-F409, Suppression Chamber N_2 Supply Isolation Valve
- o T48-F459, Bypass Leakage Isolation Valve
- o T48-F458, Pressure Control Exhaust Isolation Valve
- o T48-F457, Pressure Control Inlet Isolation Valve
- o T46-F420, Hardened Vent Inboard Isolation Valve
- o T46-F421, Hardened Vent Outboard Isolation Valve

==========================CAUTION===============================
Performance of the following step will release radioactive gas/steam to the Reactor Building fifth floor.
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7.9 **IF** primary containment pressure cannot be reduced,
THEN open T46-F410, Air Inlet Refueling Area Isolation Valve

7.10 **WHEN** primary containment pressure is reduced to 40 to 45 psig,
THEN close or verify closed:

1. T46-F400, Suppression Chamber Purge Isolation Valve
- o T46-F401, SGTS Suppression Chamber Purge Line Isolation Valve
- o T46-F412, SGTS Suppression Chamber Purge Line Isolation Bypass Valve
- o T48-F404, Suppression Chamber Inlet Isolation Valve
- o T48-F409, Suppression Chamber N_2 Supply Isolation Valve
- o T48-F459, Bypass Leakage Isolation Valve
- o T48-F458, Pressure Control Exhaust Isolation Valve
- o T48-F457, Pressure Control Inlet Isolation Valve
- o T46-F410, Air Inlet Refueling Area Isolation Valve
- o T46-F420, Hardened Vent Inboard Isolation Valve
- o T46-F421, Hardened Vent Outboard Isolation Valve

7.11 Restore SGTS to operation in accordance with Primary Containment Venting and Purge, Section 1, page 79.

7.12 **IF** T46-F406, HPCI Vacuum Pump Exhaust Isolation Valve, was closed and the HPCI System is in operation,
THEN open T46-F406, HPCI Vacuum Pump Exhaust Isolation Valve (23.202).

air and steam. The Technical Specifications delineate the limits on the use of the SGTS while purging or venting. This limit is further controlled by the Technical Specifications, which require that only one division of the SGTS be used.

- o Fermi 2 net positive suction head (NPSH) requirements for emergency core cooling system (ECCS) pumps are in conformance with Regulatory Guide 1.1. The Regulatory Guide allows no credit for positive containment pressure in the NPSH calculations. Therefore, a reduced containment pressure due to purging has no safety consequence on ECCS pump NPSH margins.

6.2.5.2.5.1 <u>Hardened Torus Vent System</u>

A hardened torus vent system has been installed at Fermi 2 under the 10 CFR 50.59 process in response to NRC Generic Letter 89-16, "Installation of Hardened Wetwell Vent".

During severe accidents which are outside the design basis, plant emergency procedures direct the operators to vent the wetwell airspace to prevent exceeding the primary containment pressure limit. Venting permits controlled releases by preventing permanent damage to the drywell. In addition, venting from the wetwell scrubs fission products from the effluent and reduces radioactive releases. The benefits of venting over a rupture of the drywell are reduced radiological consequences. The purpose of a hardened wetwell vent system is to provide a reliable design consistent with the safety objective of the plant emergency procedures.

The vent is sized to meet or exceed the BWR Owners Group (BWROG)/NRC general design criteria which require that under the
6 conditions of (1) a constant heat input at a rate equal to 1.1 percent of rated thermal power and (2) containment pressure is equal to the primary containment pressure limit (PCPL), the exhaust flow through the vent is sufficient to prevent the containment pressure from increasing.

The hardened torus vent system consists of a 10-inch, Schedule 40, carbon-steel pipe routed from the 24-inch standby gas treatment system (SGTS) inlet header on the fifth floor Reactor Building through the Reactor Building siding into a new stack which discharges at an elevated location. The 10-inch pipe contains two torus vent secondary containment fail closed isolation valves (TVSCIV). The TVSCIVs air-operated butterfly valves (AOVs) are supplied by Division II non-interruptable control air supply (NIAS). The solenoid valves are powered by the reactor protection system (RPS) and divisionally separated. The inboard AOV is powered by Division I RPS and the outboard AOV is powered by the Division II RPS. Spectacle flanges, to facilitate maintenance of the AOVs, are installed upstream and downstream of the AOVs, with one outboard spectacle flange located outside the Reactor Building. Controls and position

indications for the AOVs are located in the control room and are keylocked to prevent inadvertent positioning.

New piping from the first spectacle flange downstream from the existing header up to and including the stack is Class D, QA Level I, and Seismic Category I. This is consistent with the original classification of SGTS. The TVSCIVs maintain secondary containment integrity and are Class D, QA Level 1, Seismic Category I, and fail safe. The valves have been environmentally qualified to NUREG-0588 Category 2B (Mechanical) for pressure boundary integrity purposes. The leak tightness of the TVSCIVs is ensured by performing the secondary containment drawdown test at regular intervals.

Air supply for the primary containment isolation valves T4600F400, F401, and F412 in the SGTS has been changed from interruptable air supply (IAS) to Division 2 NIAS to improve venting reliability.

The pilot solenoid valves for the TVSCIVs are supplied by NIAS and are Class D, QA Level I, Seismic Category I, and have been environmentally qualified to NUREG-0588 Category 2B (Mechanical) and 2C (Electrical) to maintain the pressure boundary integrity of NIAS. The limit switches are QA Level non-Q, Seismic Category II/I.

A radiation monitor has been installed on the new 10-inch pipe to enable monitoring of any radiological releases when the vent is open. The monitor is powered from Division I RPS. The monitor is QA Level non-Q, Seismic Category II/I, and has indication and alarm in the control center to alert the operators of a radiological release. The monitor also has an interface with the Emergency Response Information System (ERIS). Arrangement details are shown in Figure 11.4-4. The details of the radiation monitoring system are described in Subsection 11.4.3.11.3. 6

The torus hardened vent system components which require electrical power are the radiation monitor, solenoid valves, and the controls of the hardened vent air operated isolation valves. There are two TVSCI valves in series that are keylock switch controlled and fail closed. To preclude any inadvertent opening of the vent line to the atmosphere and jeopardizing secondary containment integrity due to a single failure, the two TVSCIV pilot solenoid valves are powered by difference Divisions of RPS. The radiation monitor is powered from Division I of RPS.

The hardened vent system is designed to be used for events that are outside the design basis of the plant. Therefore, the system does not comply with the design basis described in Subsection 6.2.5.1. The RPS power supply is selected to power the above components for reliable operation of the system. The RPS branch circuits feeding the hardened vent system components are adequately protected through properly coordinated safety grade fuses. Since RPS is a fail-safe system and the branch circuits used in the hardened vent system are properly protected, any

single failure in the hardened vent system cannot prevent the
RPS' ability to scram the reactor when it is needed. The power
supply to each of these valves is divisionally separated and each
valve control circuit is defeated through a normally open contact
of a qualified keylocked selector switch; thus no single failure
6 can inadvertently open the vent path nor can it prevent the
ability of the RPS system from performing the scram action when
it is needed. Furthermore, the RPS power to non-safety grade
torus hardened vent system components is consistent with Fermi 2
design practices and by design any potential of full scram due to
single failure or non-Q component failure in the hardened vent
system is avoided. Therefore, the RPS system's intended design
function to safely shut down the reactor is not compromised.

6.2.5.3 Safety Evaluation

The safety evaluation of the CGC system is considered in terms of the design of the system to perform its intended function. This evaluation depends on the hydrogen generation analysis in the primary containment following the postulated LOCA and on the design of the CGC system to function.

6.2.5.3.1 Hydrogen Generation Analysis

In establishing the design and assessing the capability of the
CGC system, an analysis was performed to determine the primary
containment (drywell and torus) hydrogen and oxygen concentra-
tions as a function of time following the postulated LOCA. The
analysis was performed in accordance with the assumptions and
criteria provided in Reference 28. The results of the analysis
6 indicate that hydrogen and oxygen can be safely and effectively
controlled by the CGCS to the limit of Table 1 in Reference 28.

Plant operating procedures will prohibit recombiner initiation until containment pressure is less than 30 psia. In addition, plant procedures will indicate when the recombiner needs to be initiated.

6.2.5.3.1.1 General

Following a postulated LOCA, hydrogen gas may be generated from the following sources:

a. Metal/water reaction involving the zirconium fuel cladding and the reactor coolant

b. Radiolytic decomposition of the postaccident emergency cooling solutions

c. Corrosion of containment materials by solutions used for emergency cooling or by containment spray.