October 16, 1998

Mr. James W. Langenbach, Vice President
and Director, TMI-1
GPU Nuclear, Inc.
P.O. Box 480
Middletown, PA 17057

SUBJECT: REVIEW OF PRELIMINARY ACCIDENT SEQUENCE PRECURSOR ANALYSIS OF CONDITION AT THREE MILE ISLAND NUCLEAR STATION, UNIT NO. 1

Dear Mr. Langenbach:

Enclosed for your information is a copy of the final Accident Sequence Precursor (ASP) analysis of the operational event at the Three Mile Island Nuclear Station, Unit 1 (TMI-1) reported in Licensee Event Report (LER) Nos. 289/97-007, -008, and -010. This final analysis (Enclosure 1) was prepared by our contractor at the Oak Ridge National Laboratory (ORNL), based on review and evaluation of your comments on the preliminary analysis and comments received from our independent contractor, Sandia National Laboratories (SNL). Enclosure 2 contains our responses to your specific comments. Our review of your comments employed the criteria contained in the material which accompanied the preliminary analysis. The results of the final analysis indicate that this event is a precursor for 1997.

Please contact me at (301) 415-1402 if you have any questions regarding the enclosures. We recognize and appreciate the effort expended by you and your staff in reviewing and providing comments on the preliminary analysis.

Sincerely,

Original signed by
Timothy G. Colburn, Sr. Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-289

Enclosure: As stated

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LER Nos. 289/97-007, -008, -010

Event Description: Failure of both generator output breakers causes a LOOP

Date of Event: June 21, 1997

Plant: Three Mile Island, Unit 1

Event Summary

Three Mile Island, Unit 1 (TMI 1) was at 100% power when the plant experienced a loss of offsite power (LOOP) after both generator output breakers in the 230-kV substation failed.[1] The LOOP resulted in an immediate trip of both the reactor and the turbine; plant computer data indicated that the trip insertion times were excessive for four control rods.[2] Both emergency diesel generators (EDGs) started and loaded as designed. Offsite power was restored within 90 min. The unit was cooled by natural circulation cooling until offsite power and forced cooling were restored. It was subsequently discovered that the pressurizer power-operated relief valve (PORV) was failed closed during this event (see Additional Event-Related Information).[3] The estimated conditional core damage probability (CCDP) for this plant-centered LOOP is 9.6×10^{-6}.

Event Description

TMI 1 was at 100% power after almost 617 d of continuous operation. On June 21, 1997, the B phase of the 230-kV generator output breaker GBI-02 (Fig. 1) developed a fault causing severe overheating and the subsequent ejection of the bushing and conductor from the breaker housing. This resulted in a fault being detected on 230-kV bus 4. The parallel generator breaker, GBI-12, opened because of the detected fault on 230-kV bus 4. Breaker GBI-12 subsequently suffered a restrike, which damaged the B phase of this breaker, causing a fault on 230-kV bus 8. Automatic breaker action because of both faults isolated electric power to the station, resulting in a LOOP.[1]

The LOOP caused an immediate reactor trip and turbine trip. The plant computer captured times associated with each control rod reaching the 25% zone reference as the reactor trip occurred. A review of the data showed that four control rods exceeded the trip insertion time limit of 1.66 s for ¾ insertion. Personnel attributed the slow insertion times to reduced clearances in the old-style control rod drive thermal barriers because of the presence of deposits on the internal check valves, between the thermal barrier bushing, and on the leadscrew. All control rods inserted to the ¾ insertion position within 3.0 s. The licensee determined that there would be no adverse effects associated with control rod insertion times as high as 3.0 s (Ref. 2).

Both EDGs started and loaded onto their respective safeguards bus as designed. Nonvital loads, including main feedwater, condensate, circulating water, and main condenser vacuum pumps, were not energized. The reactor coolant pumps were also without power. Natural circulation was verified in the reactor coolant system within 19 min following the trip and LOOP. Decay heat removal was established using the emergency feedwater (EFW) system and the steam generator atmospheric dump valves. Offsite power was restored within 90 min

Enclosure 1

after the breaker failures. After operators established the main condenser heat sink, the reactor coolant pumps were restarted. The reactor coolant system was returned to forced circulation cooling ~9 h after the unit tripped.[1]

Additional Event-Related Information

TMI 1 has a single PORV installed on the pressurizer that is replaced with a spare PORV during each refueling outage. The licensee discovered that during the previous refueling outage, the PORV was wired incorrectly and was subsequently inoperable for the entire operating cycle. The PORV was failed in the closed position and would not have opened in response to an automatic (2450 psig) or a manual signal.[3] The operating cycle completed with a failed PORV encompassed the LOOP event described by Refs. 1 and 2.

Besides the PORV, there are two safety relief valves connected to the pressurizer with a nominal relief setpoint of 2500 psig. The shutoff head of the safety injection pumps is ~2900 psig. Feed-and-bleed operation is possible with an inoperable PORV because the safety relief valves can be lifted with the head established by operating the safety injection pumps.[4]

TMI 1 has two dedicated EDGs (1A and 1B) to supply electric power to engineered safeguards buses 1D and 1E, respectively, in the event of a LOOP. Additionally, one EDG previously from TMI 2 is available as an alternate ac power source during a station blackout (SBO). The alternate EDG, which is manually started from the control room, can be aligned to either engineered safeguards bus 1D or 1E or the balance-of-plant bus 1C within 10 min following an SBO. Operators must close two breakers and open/lockout two breakers, and any desired loads must be manually loaded onto the bus selected to be reenergized.[4]

Modeling Assumptions

This event was modeled as a plant-centered LOOP. The probability of not recovering offsite power in the short term is included in the initiating event probability (IE-LOOP). That is, the probability of a LOOP is 1.0. The probability that offsite power is not recovered in ~30 min is 0.5, based on the data distributions provided in NUREG-1032, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*.[5] Consequently, IE-LOOP was set to the probability for a plant-centered LOOP assuming operators fail to recover offsite power in the short term (5.0×10^{-1}).

The probability of short-term and long-term offsite power recovery for a plant-centered LOOP and the probability of a reactor coolant pump (RCP) seal loss-of-coolant accident (LOCA) following a postulated station blackout were developed based on data distributions contained in NUREG-1032. The RCP seal LOCA models were developed as part of the NUREG-1150 probabilistic risk assessment (PRA) efforts. Both models are described in *Revised LOOP Recovery and PWR Seal LOCA Models*.[6] The probabilities for the following basic events are based on these models:

1. initiating event–LOOP (IE-LOOP),
2. operator fails to recover offsite power within 2 h (OEP-XHE-NOREC-2H),
3. operator fails to recover offsite power within 6 h (OEP-XHE-NOREC-6H),

4. operator fails to recover offsite power before battery depletion (OPE-XHE-NOREC-BD),
5. operator fails to recover offsite power before RCP seals fail (OPE-XHE-NOREC-SL), and
6. RCP seals fail without cooling and injection water (RCS-MDP-LK-SEALS).

The PRA for TMI 1 indicates that an SBO with a concurrent failure of EFW would lead to core damage in approximately 2 h (Ref. 7, Table B.1-11, page B.1-25). This indicates that substantial time is available for the recovery of electric power. Potential recovery actions were modeled (using NUREG-1032) by the addition of a basic event (OEP-XHE-NOREC-SB) under the OP-SBO top event (OP-2H) on the LOOP event tree (Fig. 2). Top event OP-SBO is substituted for the OP-2H top event whenever emergency power (EP) and EFW fail.

The alternate EDG (from TMI 2) was added to the Integrated Reliability and Risk Analysis System (IRRAS) model for TMI 1. The probability that the alternate EDG fails to start and run (basic event EPS-DGN-FC-AAC) was set to the same value as the dedicated EDGs (4.2×10^{-2}). In addition, because operators must start and load the alternate EDG manually, a basic event was added to reflect the probability that the operator fails to start and load the alternate EDG (basic event EPS-XHE-XM-AAC). Basic event EPS-XHE-XM-AAC was set at 1.0×10^{-2} in accordance with similar human error probabilities already incorporated in the IRRAS model for TMI. The common-cause failure probability of the emergency power system for the base case was based on two EDGs. This was adjusted based on the availability of three EDGs and was developed based on data distributions contained in INEL-94-0064, *Common-Cause Failure Data Collection and Analysis System* (Ref. 8, Table 5-8: alpha factor distribution summary – fail to start, CCCG = 3, $\alpha_{3S} = 0.0224$; and Table 5-11: alpha factor distribution summary – fail to run, CCCG = 3, $\alpha_{3R} = 0.0232$). Because α_3 is equivalent to the β factor of the multiple Greek letter method used in the IRRAS models, the common-cause failure probability of the EDGs (basic event EPS-DGN-CF-ALL) was adjusted from 1.6×10^{-3} based on two EDGs to 9.5×10^{-4} based on three EDGs.

The slow insertion of four reactor control rods was not considered in the model. The control rods inserted well within the time (3.0 s) that the licensee calculated to be limiting. Additionally, all but four control rods met the ¾ insertion time prescribed by the Technical Specifications.

Because the PORV was inadvertently disabled during the operating cycle that encompassed the LOOP event, the probability that the PORV fails to open on demand (basic event PPR-SRV-CC-PORV) was set to "TRUE" (i.e., will not open). Two additional basic events were added to the IRRAS model to account for the availability of the safety relief valves to relieve any pressure buildup (basic events PPR-SRV-CC-1A and PPR-SRV-CC-1B). The probability that a safety relief valve would fail to open when its setpoint was reached was set to the nominal failure rate for the PORV (see Table 1). Additionally, the operator would only need to verify that the high-pressure injection (HPI) pumps started in a situation that required feed-and-bleed cooling to remove decay heat; no other action regarding the PORV or the safety relief valves is required from the operators. Therefore, the probability that the operator fails to initiate feed-and-bleed cooling (basic event HPI-XHE-XM-HPICL) was reduced from 1.0×10^{-2} to 1.0×10^{-3}.

Analysis Results

The CCDP for this event is 9.6×10^{-6}. The dominant core damage sequence for this event (sequence 26 on Fig. 2) involves

- a LOOP,
- a successful reactor trip,
- a failure of emergency power,
- a successful initiation of emergency feedwater,
- no challenge to the PORV (failed) or pressurizer safety relief valves,
- a failure of the reactor coolant pump seals, and
- a failure to restore electric power before core damage.

This SBO sequence (sequence 26 on Fig. 2) accounts for 85% of the total contribution to the CCDP. The next most dominant sequence (sequence 41 on Fig. 2) contributes 11% to the total CCDP. This sequence involves an SBO, a failure of the EFW system, and a failure to recover any form of electrical power before the onset of core damage.

All of the most significant sequences involve an SBO. The nominal probability that a PORV is challenged during a LOOP or an SBO is 0.16 and 0.37, respectively, based on actual LOOP and SBO events. However, the PORV failure diminished those sequences where the PORV could potentially have lifted and then failed to reseat. Basic events PPR-SRV-CO-L and PPR-SRV-CO-SBO (defined in Table 1) were set to 5.0×10^{-3} and 5.4×10^{-3}, respectively (i.e., the PORV will not open, but the safety valves are challenged) based on Integrated Plant Examination (IPE) data. Because the safety relief valves have a higher set point than the PORV, they may or may not have lifted in place of the PORV. However, the safety valves were assumed to lift during feed-and-bleed operation.

Definitions and probabilities for selected basic events are shown in Table 1. The conditional probabilities associated with the highest probability sequences are shown in Table 2. Table 3 lists the sequence logic associated with the sequences listed in Table 2. Table 4 describes the system names associated with the dominant sequences. Minimal cut sets associated with the dominant sequences are shown in Table 5.

Acronyms

CCDP	conditional core damage probability
CDP	core damage probability
EDG	emergency diesel generator
EFW	emergency feedwater system
HPI	high-pressure injection
IPE	integrated plant examination
IRRAS	Integrated Reliability and Risk Analysis System
kV	kilo-Volt
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
MOV	motor-operated valve

PORV	power-operated relief valve
PRA	probabilistic risk assessment
RCP	reactor coolant pump
SBO	station blackout
SGTR	steam generator tube rupture
SLOCA	small loss-of-coolant accident
TMI 1	Three Mile Island, Unit 1
TRANS	transient event

References

1. LER 289/97-007, Rev. 0, "Generator Output Breaker Failure Resulting in a Loss of Off-Site Power and Reactor Trip," July 21, 1997.

2. LER 289/97-008, Rev. 0, "Control Rod Trip Insertion Times Exceed TS Section 4.7.1.1 Limits," July 21, 1997.

3. LER 289/97-010, Rev. 0, "Pilot Operated Relief Valve (PORV) Inoperability Due to Being Mis-Wired and Failure to Perform Post-Maintenance Test (PMT) Following Replacement During 11R Refueling Outage," November 12, 1997.

4. Three Mile Island, *Final Safety Analysis Report (Updated Version)*.

5. P. W. Baranowsky, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032, U. S. Nuclear Regulatory Commission, June 1988.

6. *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989.

7. Three Mile Island Unit 1, *Probabilistic Risk Assessment (Level 1)*, December 1992.

8. Marshall and Rasmuson, *Common-Cause Failure Data Collection and Analysis System*, INEL-94/0064, December 1995.

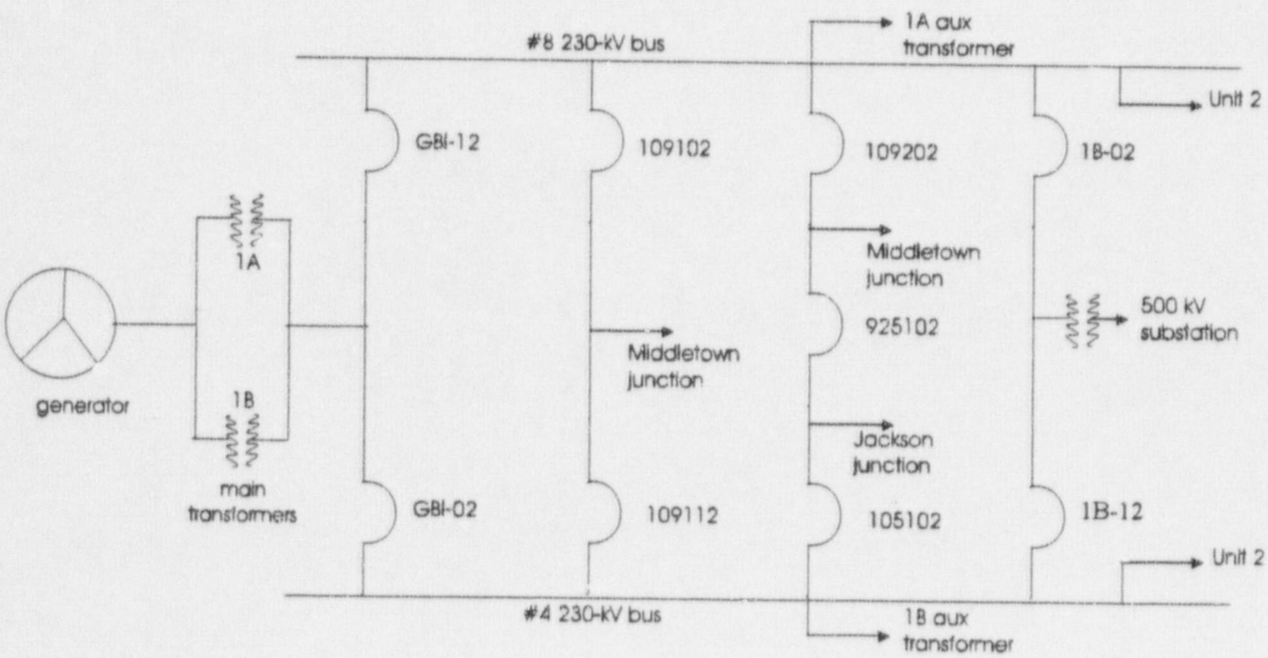

Fig. 1 Three Mile Island switchyard.

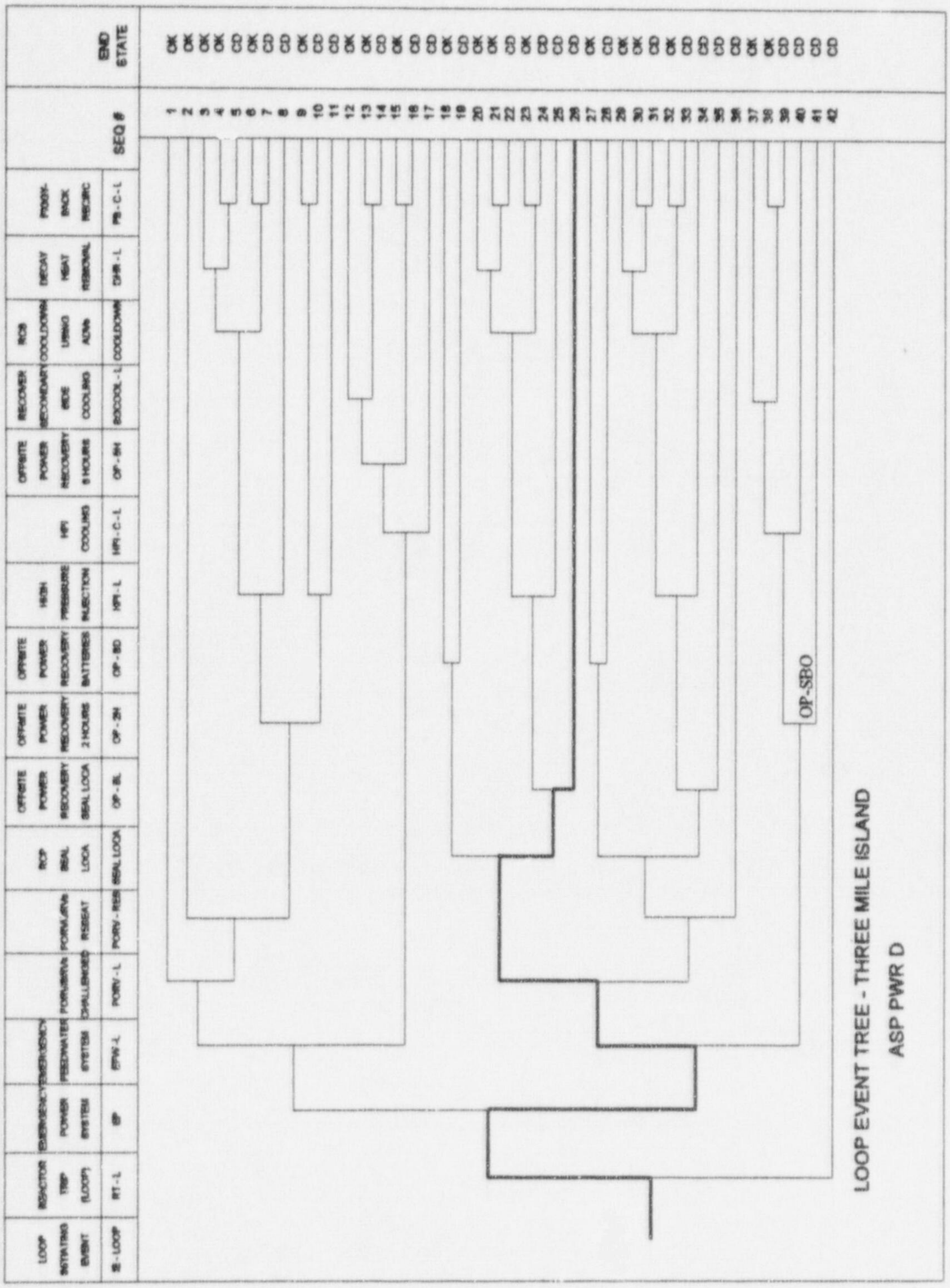

Fig. 2 Dominant core damage sequence for LER Nos. 289/97-007, -008, -010.

Table 1. Definitions and Probabilities for Selected Basic Events for LER Nos. 289/97-007, -008, - 010

Event name	Description	Base probability	Current probability	Type	Modified for this event
IE-LOOP	Initiating Event–LOOP (Includes the Probability of Recovering Offsite Power in the Short Term)	8.6 E-006	5.0 E-001		Yes
IE-SGTR	Initiating Event–Steam Generator Tube Rupture	1.6 E-006	0.0 E+000		Yes
IE-SLOCA	Initiating Event–Small Loss-of-Coolant Accident (SLOCA)	1.0 E-006	0.0 E+000		Yes
IE-TRANS	Initiating Event–Transient (TRANS)	1.3 E-004	0.0 E+000		Yes
EFW-TDP-FC-TDP	EFW Turbine-Driven Pump Fails	3.2 E-002	3.2 E-002		No
EFW-XHE-NOREC-EP	Operator Fails to Recover EFW during an SBO	3.4 E-001	3.4 E-001		No
EPS-DGN-CF-ALL	Common-Cause Failure of EDGs	9.5 E-004	9.5 E-004		No
EPS-DGN-FC-1A	1A EDG Fails to Start and Run	4.2 E-002	4.2 E-002		No
EPS-DGN-FC-1B	1B EDG Fails to Start and Run	4.2 E-002	4.2 E-002		No
EPS-DGN-FC-AAC	Alternate ac EDG Fails to Start and Run	4.2 E-002	4.2 E-002	NEW	No
EPS-XHE-NOREC	Operator Fails to Recover Emergency Power	8.0 E-001	8.0 E-001		No
EPS-XHE-XM-AAC	Operator Fails to Start or Load the Alternate ac EDG	1.0 E-002	1.0 E-002	NEW	No
OEP-XHE-NOREC-2H	Operator Fails to Recover Offsite Power within 2 h	2.2 E-001	1.4 E-001		Yes
OEP-XHE-NOREC-6H	Operator Fails to Recover Offsite Power within 6 h	6.7 E-002	9.9 E-004		Yes
OEP-XHE-NOREC-BD	Operator Fails to Recover Offsite Power before Battery Depletion	2.4 E-002	3.5 E-004		Yes
OEP-XHE-NOREC-SB	Operator Fails to Recover Electric Power before Core Damage (No Electric Power or EFW)	2.3 E-001	2.3 E-001	NEW	No
OEP-XHE-NOREC-SL	Operator Fails to Recover Offsite Power before RCP Seals Fail	5.7 E-001	4.8 E-001		Yes

Table 1. Definitions and Probabilities for Selected Basic Events for LER Nos. 289/97-007, -008, -010 (Continued)

Event name	Description	Base probability	Current probability	Type	Modified for this event
PPR-SRV-CC-1A	Safety Relief Valve 1A Fails to Open on Demand	6.3 E-003	6.3 E-003	NEW	No
PPR-SRV-CC-1B	Safety Relief Valve 1B Fails to Open on Demand	6.3 E-003	6.3 E-003	NEW	No
PPR-SRV-CC-PORV	PORV Fails to Open on Demand	6.3 E-003	1.0 E+000	TRUE	Yes
PPR-SRV-CO-L	PORV Fails to Open, but Safety Valves Challenged during a LOOP	1.6 E-001	5.0 E-003		Yes
PPR-SRV-OO-SBO	PORV Fails to Open, but Safety Valves Challenged during an SBO	3.7 E-001	5.4 E-003		Yes
RCS-MDP-LK-SEALS	RCP Seals Fail without Cooling and Injection	4.6 E-002	4.0 E-002		Yes

Table 2. Sequence Conditional Probabilities for LER Nos. 289/97-007, -008, -010

Event tree name	Sequence number	Conditional core damage probability (CCDP)	Percent contribution
LOOP	26	8.2 E-006	85.0
LOOP	41	1.1 E-006	10.8
LOOP	19	1.5 E-007	1.5
Total (all sequences)		**9.6 E-006**	

Table 3. Sequence Logic for Dominant Sequences for LER Nos. 289/97-007, -008, -010

Event tree name	Sequence number	Logic
LOOP	26	/RT-L, EP, /EFW-L, /PORV-SBO, SEALLOCA, OP-SL
LOOP	41	/RT-L, EP, EFW-EP, OP-SBO
LOOP	19	/RT-L, EP, /EFW-L, /PORV-SBO, /SEALLOCA, OP-BD

Table 4. System Names for LER Nos. 289/97-007, -008, -010

System name	Logic
EFW-EP	No or Insufficient EFW Flow during an SBO
EFW-L	No or Insufficient EFW Flow during a LOOP
EP	Failure of Both Trains of Emergency Power
OP-BD	Operator Fails to Recover Off-Site Power before Battery Depletion
OP-SBO	Operator Fails to Restore ac Power before Core Damage Occurs Following an SBO and Loss of EFW
OP-SL	Operator Fails to Restore ac Power before a Reactor Coolant Pump Seal LOCA Occurs
PORV-SBO	PORV/Safety Relief Valves Challenged during an SBO
RT-L	Reactor Fails to Trip during a LOOP
SEALLOCA	Reactor Coolant Pump Seals Fail during a LOOP

Table 5. Conditional Cut Sets for Higher Probability Sequences for LER Nos. 289/97-007, -008, -010

Cut set number	Percent contribution	CCDP[a]	Cut sets
LOOP Sequence 26		8.2 E-006	
1	91.2	7.5 E-006	EPS-DGN-CF-ALL, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
2	7.1	5.8 E-007	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-DGN-FC-AAC, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
3	1.7	1.4 E-007	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-XM-AAC, EPS-XHE-NOREC, RCS-MDP-LK-SEALS, OEP-XHE-NOREC-SL
LOOP Sequence 41		1.1 E-006	
1	91.1	9.6 E-007	EPS-DGN-CF-ALL, EPS-XHE-NOREC, EFW-TDP-FC-TDP, EFW-XHE-NOREC-EP, OEP-XHE-NOREC-SB
2	7.1	7.5 E-008	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-DGN-FC-AAC, EPS-XHE-NOREC, EFW-TDP-FC-TDP, EFW-XHE-NOREC-EP, OEP-XHE-NOREC-SB
3	1.7	1.8 E-008	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-XM-AAC, EPS-XHE-NOREC, EFW-TDP-FC-TDP, EFW-XHE-NOREC-EP, OEP-XHE-NOREC-SB
LOOP Sequence 19		1.5 E-007	
1	91.2	1.3 E-007	EPS-DGN-CF-ALL, EPS-XHE-NOREC, OEP-XHE-NOREC-BD
2	7.1	1.0 E-008	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-DGN-FC-AAC, EPS-XHE-NOREC, OEP-XHE-NOREC-BD
3	1.7	2.5 E-009	EPS-DGN-FC-1A, EPS-DGN-FC-1B, EPS-XHE-XM-AAC, EPS-XHE-NOREC, OEP-XHE-NOREC-BD
Total (all sequences)		9.6 E-006	

[a]The conditional probability for each cut set is determined by multiplying the probability of the initiating event by the probabilities of the basic events in that minimal cut set. The probabilities for the initiating events and the basic events are given in Table 1.

LER No. 289/97-007, -008, -010

Event Description: Failure of both generator output breakers causes a LOOP

Date of Event: June 21, 1997

Plant: Three Mile Island, Unit 1

Licensee Comments

Reference: Letter from James W. Langenbach, Vice President and Director, TMI, to U. S. Nuclear Regulatory Commission, "GPU Nuclear Review of Preliminary Accident Sequence Precursor Analysis of Operational Event at Three Mile Island Nuclear Station, Unit No. 1," May 1, 1998.

Comment 1a: Page 1, under **Event Summary**, the fifth sentence states, "The unit was cooled by natural circulation cooling until offsite power was restored." This sentence should state, "The unit was cooled by natural circulation cooling until offsite power and forced cooling were restored." Per LER 97-009 pages 4 and 5, it took 1.5 h for offsite power to be restored. Another 7.5 h passed before forced cooling (RC-P-1A/B/C/D) was restored.

Response 1a: The proposed editorial change was made. This has no effect on the analysis results, but more accurately reflects the sequence of events depicted in the time line outlined in LER 97-009, pages 4 and 5.

Comment 1b: Page 1, under **Event Description**, first paragraph, the second sentence states, "On June 21, 1997, the B phase of the 230-kV power transformer developed a fault causing severe overheating and the subsequent ejection of the bushing and conductor from the breaker housing of output breaker GBI-02 (Fig. 1)." This sentence should state "On June 21, 1997, the B phase of the 230-kV generator output breaker GBI-02 (Fig. 1) developed a fault causing severe overheating and the subsequent ejection of the bushing and conductor from the breaker housing." Per LER 97-007, there is no evidence that the fault location was in the power transformer.

Response 1b: The proposed editorial change was made. This has no effect on the analysis results.

Enclosure 2

Comment 1c: Page 1, under **Event Description**, second paragraph, second to the last sentence states, "All control rods inserted to ¾ insertion position within 2.2 s." This sentence should state, "All control rods inserted to the ¾ insertion position well within 3.0 s." Per LER 97-008 there is no claim made that at the time of the reactor trip that all control rods inserted to the ¾ position within 2.2 s. The only claim is that retested rods came in at less than 2.2 s. The LER does indicate that the rods did insert to the ¾ position well within 3.0 s at the time of the reactor trip.

Response 1c: The sentence was changed to, "All control rods inserted to the ¾ insertion position within 3.0 s." LER 97-008 only indicates that rod insertion times were below 3.0 s without quantifying any specific rod drop times during the reactor trip. Therefore, the times were not categorized as being "well below 3.0 s." This has no effect on the analysis results.

Comment 1d: Page 1, under **Event Description**, third paragraph, second sentence states, "Nonvital loads, including circulating water and main condenser vacuum pumps, were not energized." This sentence should state, "Nonvital loads, including main feedwater, condensate, circulating water, and main condenser vacuum pumps, were not energized." The more important non-vital loads per LER 97-007 page 2 were omitted from the event description.

Response 1d: The proposed editorial change was made. This has no effect on the analysis results, but more accurately reflects the scope of major loads lost following the LOOP event.

Comment 1e: Page 2, under **Event Description**, last paragraph, last sentence states, "After the operators established the main condenser heat sink, the reactor coolant pumps were restarted, returning the reactor coolant system to forced circulation cooling." This should state, "After the operators established the main condenser heat sink, the reactor coolant pumps were restarted. Since the natural circulation cooling of the reactor coolant system was stable, the reactor coolant system was not returned to forced circulation cooling until nine hours after the reactor trip."

Response 1e: LER 97-007 does not indicate that the delay was based on the stability of natural circulation cooling. The last two sentences were changed to, "After operators established the main condenser heat sink, the reactor coolant pumps were restarted. The reactor coolant system was returned to forced circulation cooling ~9 h after the unit tripped." This has no effect on the analysis results.

Comment 1f: Page 6, Fig. 1; change breaker identification from 1B-14 to 1B-12 per Gilbert/ Commonwealth drawing 229-002.

Response 1f: The proposed figure change was made. This has no effect on the analysis results.

Comment 2a: Page 2, under **Additional Event-Related Information**, the first and second paragraphs are considered to place too much emphasis on the PORV being inoperable. The reactor coolant system pressure did not approach the PORV set point at any time during the LOOP event. Therefore, the PORV being inoperable was not a factor during the LOOP event.

Response 2a: The ASP Program deals with the probability that the PORV set point may be reached following any LOOP event or that feed-and-bleed cooling operations may be required if emergency feedwater were to fail. In a generic analysis, cut sets would exist in the analysis output indicating that the PORV had opened. However, to correctly model this specific event, the basic event governing the PORV opening needed to be set to TRUE (would not open). Some discussion was required to establish the basis for this basic event change. Because these two paragraphs provide essentially all of the information on the inoperable PORV (there is one sentence under **Modeling Assumptions**), no change in the PORV discussion was made in the analysis.

Comment 2b: Page 2, under **Additional Event-Related Information**, last paragraph, the second and third sentences state, "Additionally one EDG from TMI 2 is available as an alternate ac power source during a station blackout (SBO). The alternate EDG, which is manually started from the control room, can be aligned to either engineered safeguards bus 1D or 1E within 10 min following an SBO." These sentences more correctly should state, "Additionally, one EDG previously from TMI 2 is available as an alternate ac power source during a station blackout (SBO). The alternate EDG, which is manually started from the control room, can be aligned to either engineered safeguards bus 1D or 1E or balance of plant (BOP) bus 1C within 10 min following an SBO."

Response 2b: The proposed editorial changes were made. This has no effect on the analysis results.

Comment 2c: Page 2, under **Additional Event-Related Information**, last paragraph, last sentence, states, "Operators must close two breakers and any desired loads must be manually loaded onto the bus selected to be reenergized." This sentence should state, "Operators must close two breakers and open/lockout two breakers, and any desired loads must be manually loaded onto the bus selected to be reenergized." Per abnormal procedure 1202-2, the operator must open/lockout two breakers in addition to closing two breakers to correctly align the SBO diesel with bus 1D or 1E.

Response 2c: The proposed editorial changes were made. This has no effect on the analysis results because the operator response was not considered on a breaker by breaker basis. The response was simply assumed to be a procedure-based action with a nominal probability of failure to perform the procedure of 1.0×10^{-2}, which is consistent with similar operator responses.

Comment 3a: (1) Page 3, first paragraph should give a reference for the reliability value used for OEP-XHE-NOREC-SB in Table 1 [of the preliminary analysis].

(2) The two events described in Table 1 of the preliminary ASP analysis, EPS-XHE-NOREC (Operator fails to recover emergency power) and OEP-XHE-NOREC-SL (Operator fails to recover offsite power before RCP seals fail), should be replaced by the non-recovery values REA and REC given in Table B.1-12 of the TMI IPE [Ref. 1]. REA is the nonrecovery factor for SBO sequences in which the Emergency Feedwater (EFW) operates successfully. REC is the nonrecovery factor for SBO sequences in which EFW does not operate successfully. Both REA and REC were evaluated under the assumption that at least two EDGs are recoverable.

(3) REA and REC were calculated by Monte Carlo simulation techniques as described in Sect. B.1.3.4 of the TMI IPE. Recovery of the TMI 2 EDG was not credited in the simulation. The result of the simulation shows that if EFW is unavailable, the mean time to core uncovery is 3.4 h; 10.3 h if EFW is available. The time interval of 2.1 h to core uncovery mentioned in the IPE was used as input to the STADIC code and was assumed to occur after a loss of all ac power. The STADIC model did not assume that all onsite ac power would fail at time t = 0 when offsite power is lost. In other words, the time to core uncovery was assessed in the IPE (page B.1-16) as the time of onsite power failure, plus 2.1 h.

Response 3a: (1) The text reference to the basic event OEP-XHE-NOREC-SB on page 3 of the analysis was changed to, "Potential recovery actions were modeled (using data from NUREG-1032) by the addition of a basic event (OEP-XHE-NOREC-SB) under the OP-SBO top event (OP-2H) on the LOOP event tree (Fig. 2)."

(2) In the SBO sequences, the probability of a reactor coolant pump seal LOCA and the probability of failing to recover ac power at various points in time are calculated using a convolution approach that recognizes that all probabilities are a function of time. A Weibull distribution is used to predict the LOOP-related parameters applicable for TMI as defined in ORNL/NRC/LTR-89/11 (Ref. 2). Probabilities associated with the failure to recover ac power and the potential for an RCP seal LOCA are calculated given that ac power was not restored in the short term (30 min). Additionally, the probability for the operator failure to restore emergency power is based on the assumption that the median repair time for an EDG is 4 h, as developed in NUREG-1032. The TMI IPE split fractions, REA and REC, are based on different assumptions concerning LOOP and EDG recovery as a function of time. The electric power recovery analysis described in Appendix B.1 to the TMI IPE imbedded the potential for an RCP seal loss-of-coolant accident (LOCA) within the calculation of REA and REC, and did not separately address the potential for short- and long-term recovery of ac power. Because basic events EPS-XHE-NOREC and OEP-XHE-NOREC-SL in the IRRAS model do not represent the same conditional probability that split fractions REA and REC represent, the free substitution of individual split fractions from the IPE into the IRRAS model is inappropriate without making further adjustments to the other LOOP-related parameters in the IRRAS model.

(3) The ASP IRRAS models do not currently simulate the potential for a delayed failure of EFW. EFW is assumed to essentially fail at event $t = 0$, and the time to core damage under station blackout was assumed to be 2.1 h per Sect. B.1.3.3.3 of the TMI IPE.

Comment 3b: In general, the probabilities for failure of restoration of offsite power, onsite emergency power, and reactor coolant pump seals may be reduced from those assumed in the analysis because operators have the following procedures listed below for guidance. Also, the June 21, 1997, event was successful in that offsite power was restored in 90 min, emergency onsite sources started and loaded as designed, and there was no RCP seal damage.

- Emergency Procedure 1202-2, Rev. 44, "Loss of Station Power,"which was used during the TMI 1 event, includes steps (3.10 and Attachment 4) for restoring offsite power. Because of these operator actions the probability of not restoring offsite power (IE-LOOP) should be lower than 5.0×10^{-1}.

- Emergency Procedure 1202-2, Rev. 44, "Loss of Station Power," which was used during the TMI 1 event, includes steps (3.11 and Attachment 1) for starting and loading the emergency diesel generators (EDGs), assuming the EDGs did not automatically start and load. In addition, the procedure includes steps (3.12 and Attachment 1) for starting and loading the SBO diesel assuming the Class 1E diesel(s) did not automatically start. Because of these operator actions, the probability of all diesels failing (EPS-DGN-CF-ALL) should be lower than 9.5×10^{-4}

- Emergency Procedure 1202, Rev.44, "Loss of Station Power," which was used during the TMI 1 event, includes Attachment 2, which may be used to restore reactor coolant pump seals. The restoration would be performed in a controlled manner to limit damage to the seals and pumps if injection and intermediate closed cooling water (ICCW) are lost for a long time period. Because of these operator actions, the probability of RCP seal failure (RCS-MDP-LK-SEALS) should be lower than 4.0×10^{-2}.

Response 3b: Human response is considered in the numerous system-specific operator nonrecovery probabilities within the TMI IRRAS model. The fact that procedural guidance exists is accounted for by the non-recovery probability basic events. The IE-LOOP initiating event represents the probability that ac power is not restored in the short term and is based on historical data gathered on specific categories of LOOP events. The probability of not recovering offsite power in the short term is included in the initiating event probability (IE-LOOP). That is, the probability of a LOOP is 1.0. The probability that offsite power is not recovered in ~30 min is 0.5, based on the data distributions provided in NUREG-1032, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*.[3] Consequently, IE-LOOP was set to the probability of failing to recover offsite power in the short term (5.0×10^{-1}) following a plant-centered LOOP. The conditional probability that a LOOP is not recovered in times greater than 30 min is considered in the calculation of probabilities for basic events OEP-XHE NOREC-SB, -SL, and -BD. Hence, the existence of procedural guidance impacts the historical data and subsequently the value calculated for these probabilities.

With respect to the LOOP event at TMI on June 21, 1997, success of specific systems or components does not change the probability that the LOOP could have lasted longer than 90 min and that core damage could occur. Therefore, the original analysis basic event values were not adjusted for the final report based on the existence of numerous emergency procedures.

- As noted above, the probability that offsite power is not recovered in ~30 min is 0.5, based on the data distributions provided in NUREG-1032 (Ref. 3). The existence of procedural guidance impacts the historical data and subsequently the value calculated for IE-LOOP. The IPE (Table B.1-5) indicates that the probability that offsite power is not recovered in ~30 min is 0.67, which is compatible with the value of 0.5 calculated from the data distributions in NUREG-1032.

- The common-cause failure probability of all three EDGs failing to start and run (EPS-DGN-CF-ALL) is based on data distributions contained in INEL-94-0064, *Common-Cause Failure Data Collection and Analysis System* [Ref. 4, Table 5-8: alpha factor distribution summary – fail to start, CCCG = 3, $\alpha_{3S} = 0.0224$; and Table 5-11: alpha factor distribution summary – fail to run, CCCG = 3, $\alpha_{3R} = 0.0232$]. Because α_3 is equivalent to the β factor of the multiple Greek letter method used in the IRRAS models, the common-cause failure probability of the EDGs was adjusted from 1.6×10^{-3} based on two EDGs to 9.5×10^{-4} based on two EDGs and an SBO diesel generator.

- The probability of an RCP seal LOCA following a postulated SBO event was developed based on data distributions contained in NUREG-1032 and the RCP seal LOCA models that were developed as part of the NUREG-1150 probabilistic risk assessment (PRA) efforts described in *Revised LOOP Recovery and PWR Seal LOCA Models*.[2] The probability of an RCP seal failure calculated from this source, assuming power is not restored in the short-term (30 min), is 4.0×10^{-2}.

Comment 4: Based on the variation in the combination of events and modeling assumptions used by the NRC in the analysis of the operational event at TMI as described in the preliminary ASP analysis, the estimate of the CCDP of the LOOP was determined to be 9.6×10^{-6}. Using the combination of events and modeling assumptions of the TMI 1 PRA (Level 1) Update, December 1992, "Appendix B – Special Analysis", which gives consideration to both the REA and REC factors, GPU Nuclear estimates the CCDP of the LOOP to be 9.53×10^{-7}.

Response 4: Numerous basic events in the IRRAS model are based on the conditional probability that an ac power source is not reestablished within 30 min. This is the reason, for example, that the initiating event probability (IE-LOOP) is 0.5 although a LOOP actually occurred. The TMI IPE incorporates these time conditional probabilities into split fractions REA and REC. IE-LOOP would have to be adjusted to 1.0 to accommodate the use of split fractions REA and REC. Adjusting IE-LOOP would subsequently require numerous other basic events to be modified.

As discussed in the response to Comment 3a, the use of a mix of basic event probabilities from the TMI IPE and IRRAS models is not appropriate without accounting for the underlying assumptions associated with substituted values. Collapsing the ASP model SBO sequences to two scenarios, one with EFW available and one with EFW failed, and applying ac power nonrecovery probabilities equivalent to REA and REC, results in an estimated CCDP of $\sim 1.0 \times 10^{-5}$. This is the CCDP for a "nominal" LOOP using nonrecovery probabilities as described in Appendix B.1 of the TMI 1 IPE. Because it was not possible to revise REA and REC to reflect the plant-centered nature of the LOOP, it was not possible to directly compare the impact of REA and REC if they were used in the analysis. Since a plant-centered LOOP is more easily recovered than a nominal LOOP, the associated CCDP would be expected to be lower than the estimate for a nominal LOOP–typically by a factor of 2–3. Hence, the plant-centered LOOP at TMI using TMI-developed nonrecovery distributions (from App. B.1 of the TMI IPE) should be in the mid-10^{-6} range. The original analysis result, which calculated a CCDP of 9.6×10^{-6}, appears to be a reasonable estimate of the importance of this event.

References:

1. Three Mile Island Unit 1, *Probabilistic Risk Assessment (Level 1)*, December 1992.

2. *Revised LOOP Recovery and PWR Seal LOCA Models*, ORNL/NRC/LTR-89/11, August 1989.

3. P. W. Baranowsky, *Evaluation of Station Blackout Accidents at Nuclear Power Plants*, NUREG-1032, U. S. Nuclear Regulatory Commission, June 1988.

4. Marshall and Rasmuson, *Common-Cause Failure Data Collection and Analysis System*, INEL-94/0064, December 1995.