May 24, 1993

Docket No. 50-255

Consumers Power Company
ATTN: Mr. Gerald B. Slade
General Manager
Palisades Nuclear Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

Dear Mr. Slade:

SUBJECT: NRC INSPECTION REPORT NO. 50-255/93010(DRS)

This refers to the inspection conducted by Messrs. J. Walker and W. Pegg of this office on April 6 through May 4, 1993. The inspection included a review of authorized activities for your Palisades Nuclear Plant. This inspection focused on the implementation of procedural changes and training modifications described by NRC Safety Evaluation Report (SER) dated February 28, 1986, regarding the single failure issue for main steam isolation valves (MSIVs). The operability and maintenance history of the MSIVs were also examined. At the conclusion of the inspection, the findings were discussed with those members of your staff identified in the enclosed report.

Areas examined during the inspection are identified in the report. The inspectors selectively examined procedures and representative records, made observations, conducted interviews with your personnel, and conducted two simulator runs of a dual steam generator blowdown concurrent with a failure of the intact steam generator's MSIV to close with operations personnel.

No violations of NRC requirements were identified during the course of the inspection. However, certain other activities, set forth in the enclosure to this letter, appear to be a deviation from commitments which you have made in previous correspondence with the Commission. A written response is required.

In addition, certain actions taken in response to the 1986 SER did not fully address the concerns expressed by the NRC. These actions should be reexamined to ensure the intent of the SER has been met. These are considered open items. A written response is requested within 60 days of the date of this report to address the open items identified in this report.

In accordance with 10 CFR 2.790 of the Commission's regulations, a copy of this letter, the enclosures, and your responses to the letter will be placed in the NRC Public Document Room.

The responses directed by this letter and the accompanying Notice are not subject to the clearance procedures of the Office of Management and Budget as required by the Paperwork Reduction Act of 1980, PL 96-511.

IE01

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

original signed by

T. O. Martin, Acting Director
Division of Reactor Safety

Enclosures:
1. Notice of Deviation
2. Inspection Report
   No. 50-255/93010(DRS)

cc w/enclosures:
David P. Hoffman, Vice President
  Nuclear Operations
OC/LFDCB
Resident Inspector, RIII
James R. Padgett, Michigan Public
  Service Commission
Michigan Department of
  Public Health
A. H. Hsia, LPM, NRR
SRI, Big Rock Point
W. Hodges, RI
A. Gibson, RII
S. Collins, RIV
K. Perkins, RV

bcc w/enclosures: PUBLIC-IE01

SEE PREVIOUS CONCURRENCE PAGE

RIII	RIII	RIII	RIII	RIII	RIII
Pegg/jw/wp/cg	Walker	Burdick	Jorgensen	Ring	Martin
05/ /93	05/ /93	05/ /93	05/ /93	05/24/93	05/24/93

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T. O. Martin, Acting Director
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RIII	RIII	RIII	RIII	RIII	RIII
Pegg/jw/wp/cg	Walker	Burdick	Jorgensen	Ring	Martin
05/21/93	05/21/93	05/21/93	05/21/93	05/ /93	05/ /93

NOTICE OF DEVIATION

Consumers Power Company
Palisades Nuclear Plant

Docket No. 50-255
License No. DPR-20

During an NRC inspection conducted on April 6 through May 4, 1993, a deviation of your written response to a safety evaluation report (SER) dated February 28, 1986, regarding the single failure issue for main steam isolation valves was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Action", 10 CFR Part 2, Appendix C, the deviation is listed below:

In an April 28, 1986 letter to the NRC, the licensee committed, in part, to perform certain actions for a main steam line break coincident with failure of the intact steam generator's main steam isolation valve to close. These specific actions include opening the power-operated relief valves, maximizing safety injection flow (feed and bleed), and maximizing auxiliary feed flow to one steam generator if the decision is made not to rely on any in-containment instrumentation.

In the basis document for emergency operating procedures (EOP) 9.0, "Functional Recovery Procedure," and EOP 6.0, "Excessive Steam Demand Event," the licensee made the conservative assumption that no in-containment instrumentation would remain operable.

Contrary to the above, as of May 4, 1993, the emergency operating procedures do not decisively instruct the operators to perform the above actions. Instead, EOP 9.0 ambiguously directs the operators to monitor in-containment parameters, which may be inoperable or unreliable, to determine the proper mitigating procedure.

Please provide to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D. C. 20555, with a copy to the Regional Administrator, Region III, in writing within 30 days of the date of this Notice, the reason for the deviation, the corrective steps which have been taken and the results achieved, the corrective steps which will be taken to avoid further deviations, and the date when your corrective action will be completed. Where good cause is shown, consideration will be given to extending the response time.

Dated at Glen Ellyn, Illinois
this 24th day of May, 1993

T. O. Martin, Acting Director
Division of Reactor Safety

U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-255/93010(DRS)

Docket No. 50-255 License No. DPR-20

Licensee: Consumers Power Company
Palisades Nuclear Generating Plant
27780 Blue Star Memorial Highway
Covert, MI 49043-9530

Facility Name: Palisades Nuclear Plant

Inspection At: Covert, Michigan

Inspection Conducted: April 6 through May 4, 1993

Inspectors: T Burdick for 5/24/93
J. Walker Date

T Burdick for 5/24/93
W. Pegg Date

Approved By: T Burdick 5/24/93
T. Burdick, Chief Date
Operator Licensing Section 2

Inspection Summary

Inspection on April 6 through May 4, 1993 (Report No. 50-255/93010(DRS))
Areas Inspected: Special inspection to assess the implementation of commitments made in response to the 1986 SER and to assess the operability and availability of the MSIVs.
Results: Within the areas inspected, one deviation was identified regarding a conflict between the EOPs dealing with a main steam line break (MSLB) coincident with a single failure of the intact steam generator's MSIV to close and commitments made to the NRC (Section 2.4.6.1). Open items were identified regarding differences between the training of operators and the recommendations of the SER on the possible loss of instrumentation inside of containment during this event (Section 2.4.6.2); a commitment to include as-found stroke time testing of the MSIVs in the operational procedures (Section 2.4.4); operator training on loss of instrumentation and procedural changes related to this event were not developed; graphs to account for errors in pressurizer and SG narrow range level were referred to in footnotes in the safety status, but no guidance is given as to when these graphs are required (Section 2.4.7.1); and discrepancies between the containment pressure and temperature values observed during simulator runs of the MSLB event and the expected values (Section 2.4.7.3). No problems with the maintenance or operability of the MSIVs were noted.

DETAILS

1.0 Exit Meeting Attendees

Consumers Power Company

T. Palmisano, Operations Manager
R. Orosz, Manager, Nuclear Engineering and Construction
J. Hanson, Operations Superintendent
R. Heimsath, Training Administrator
J. Kuemin, Licensing Administrator
R. Kasper, Maintenance Manager
R. Gerling, Reactor and Safety Analysis Manager
D. Malone, Operations Staff Support Supervisor
F. Yanik, Risk Assessment Supervisor
B. Kubacki, System Engineering Section Head
E. Letke, Main Steam System Engineer

U. S. Nuclear Regulatory Commission (NRC)

T. Burdick, Chief, Operator Licensing Section 2
J. Heller, Senior Resident Inspector

The NRC inspectors also contacted and interviewed other licensee personnel during the inspection.

2.0 Background Information

2.1 Purpose of the Inspection

The purpose of the inspection was to assess implementation of commitments made in response to the 1986 SER and to assess the operability and availability of the MSIVs through an examination of the MSIV maintenance and operational history.

2.2 History

In the early 1980's, it was discovered that the design of the main steam system did not preclude the possibility that a MSLB inside containment concurrent with a failure of the intact steam generator's (SG) MSIV could occur. The steam system at Palisades relies on reverse mounted check valves as MSIVs. There are no non-return check valves downstream of the MSIVs to prevent blow-down of both steam generators in the event of a steam line break inside containment concurrent with a passive failure of the opposite steam generator's MSIV to close.

In 1983, the licensee committed to make modifications to either replace the MSIVs with motor-operated valves, install non-return check valves downstream of the MSIVs to preclude backflow from one steam generator into the other on a MSLB, or qualify the instrumentation and equipment inside of containment for the bounding conditions of the dual steam generator blowdown.

In 1985, the licensee submitted a PRA to the NRC to show that the proposed modifications were not cost effective considering the probability of the event occurring and the cost of the improvements relative to the decrease in overall risk for each of the proposed modifications.

On February 28, 1986, the NRC issued an SER on this issue concurring with the licensee's evaluation provided alternative measures were implemented. These alternative measures, the licensee's response to them and the NRC's evaluation of the response are provided in Section 2.4.6 of this report.

2.3 Valve Design

The Palisades MSIVs are 30-inch Atwood-Morrill swing disc check valves that are reverse mounted and utilize an external air cylinder and piston connected to the valve disc shaft to hold the valves open. To close the MSIVs, a solenoid is actuated which vents the air from under the piston. This allows a compressed spring above the piston to expand and force the piston down, assisting the valve in closing.

2.4 Inspection Findings

2.4.1 Evaluation of Industry Experience

On September 29, 1991, while in the process of shutting down, the MSIVs at the Point Beach Nuclear Plant, Unit 1, failed to close due to corrosion problems in the pneumatic operators. On November 15, 1991, the licensee began a review of this industry experience and determined that the Palisades plant had eight Atwood-Morrill check valves of a similar design, including the MSIVs. Prior to evaluation of the MSIV failure's applicability to Palisades, the licensee's document tracking the industry experience was lost due a computer entry error and a failure of the responsible review group to return the evaluation to the originator. The review was started again on March 11, 1993, after NRC operations inspectors made inquiries about the MSIVs and the event at Point Beach. Consequently, the licensee did not complete an evaluation of the applicability of the Point Beach experience to Palisades in a time frame commensurate with the safety significance. The control exhibited over this review is considered a weakness.

2.4.2 MSIV Operational History

The inspectors reviewed licensee event reports, abnormal occurrence reports, and interviewed operations personnel to determine if the MSIVs had ever failed to close on demand. The inspectors determined that the MSIVs had failed to close three times in the period between 1972 and 1973. There have been no failures to close since 1973. The actions taken in response to the initial failures were successful in preventing recurrence.

2.4.3 MSIV Maintenance History

The inspectors reviewed work orders from the past ten years, with emphasis on maintenance activities over the past three years. In Work Order 24104695, the pneumatic operator of CV-0510 was disassembled and inspected per Procedure

MSS-M-19, "Disassembly, Inspection and Reassembly of Main Steam Isolation Valves CV-0501 and CV-0510", Revision 5. The inspection of the actuator internals revealed some spots of corrosion, scratches and other normal wear. The degree of wear and degradation discovered would not have affected the ability of the actuator to perform it's function. The licensee will disassemble and inspect the pneumatic actuator for CV-0501 during the next cold shutdown. Although provisions are made in this procedure to perform disassembly and inspection of the pneumatic actuator (step 5.3.9), step 3.8.4 allows the system engineer to bypass this step of the procedure. This step in the procedure had been bypassed since the installation of the actuators in 1979. The length of time between the installation of the pneumatic actuators in 1979 and the first disassembly and inspection of the actuators in 1992 is excessive considering the importance of the actuators and is a weakness.

The inspectors were concerned about the use of Q Sealant, Furmanite, or other temporary means to repair stuffing box and packing leaks without the benefit of as-found testing to determine if such activities unduly affected the performance of the MSIVs. As-found testing would provide necessary feedback on the impact of such repairs on the shaft friction and the ability of the actuator to overcome the increased friction.

The inspectors found the maintenance procedures to be clear and complete. The inspectors interviewed maintenance personnel and determined that the maintenance personnel were qualified and cognizant of the maintenance requirements of the MSIVs. No other problems were noted.

2.4.4 MSIV Testing

The inspectors reviewed the technical specification (TS) surveillance history of the MSIVs for the past five years. The MSIVs closed within the TS allowable times and conditions. Procedure RI-17, "Main Steam Isolation Valve Circuits Test and Valve Closure Timing", Revision 12, documented the required TS closure time testing. The required testing, however, did not accurately reflect the ability of the MSIVs to perform their safety function during the previous operational cycle, as the MSIVs were tested on power ascension after they had already been exercised, repaired or conditioned.

Prior to this inspection, the licensee had documented closure of the MSIVs on a shut down check list, but no measurement of the closure time was taken. The licensee has since committed to include as-found stroke time testing of the MSIVs in the operational procedures. The results of this as-found testing will indicate if there was any degradation in the ability of the MSIVs to perform their function over the previous operational cycle. The commitment to include as-found stroke time testing of the MSIVs in the operation procedures will be tracked as an open item (255/93010-01(DRS)).

2.4.5 Dual SG Blowdown

The following is a brief synopsis of what would happen in a dual SG blowdown, based on the PRA and simulator runs of this scenario on the Palisades certified plant specific simulator. The use of the designators "A" and "B" are arbitrary and are provided for clarity.

- A guillotine break occurs on the B Main Steam line, upstream of the MSIV.

- The MSIVs receive a low steam generator pressure signal to close.

- The MSIV for the A steam generator fails to close and remains in the full open position.

- Backflow from the A steam generator pushes the MSIV for the B steam generator open and allows the A steam generator to blow down through the break.

- Both steam generators blow down into containment.

- The primary coolant system (PCS) cools down to approximately 350°F and depressurizes to approximately 1000 psia. All primary coolant pumps (PCPs) are stopped.

- Upon dry out of the "Most Affected" steam generator the PCS begins to reheat and repressurize. The saturation temperature in the "Least Affected" steam generator is initially above PCS temperature preventing it from acting as a heat sink and inhibiting natural circulation.

- Voids are formed in the upper head region of the reactor vessel, again inhibiting natural circulation flow.

2.4.6 Licensee Response to SER

2.4.6.1 Operator Actions

The NRC concern, documented in the SER issued in 1986, was as follows:

> "The emergency procedures dealing with secondary line breaks, EOP-6 and EOP-7, and the procedure for normal reactor trip, EOP-1, do not provide definitive guidance regarding maintenance of a heat sink (use of steam generator level or other secondary parameter for feedback) even if the operator perceives that both steam generators may be faulted. In fact, in EOP-1 it is noted that if dry out occurs, the affected steam generator is to be considered inoperable. Additionally, the absence in the procedures of any recognition of overcooling expected to occur in these events enhances the potential for inappropriate spontaneous operator action. Procedures based on systems analysis should be developed to enable the operator to cope with these events and to achieve a controlled cooldown. The staff considers that analyses are needed to identify a control strategy that considers the decay heat generation available to the operator and how the response changes over the course of the cooldown."

The licensee's response to this concern, documented in a letter to the NRC dated April 28, 1986, was as follows:

> "All new EOPs provide definitive guidance regarding maintenance of the heat sink. This includes monitoring of specific parameters including wide range steam generator levels, core temperature indications, steam generator pressures and steam generator feed flow. All new EOPs have safety function status checks which include the safety function of core heat removal and PCS heat removal."
>
> "In all new EOPs, emphasis is placed on continuous feed to at least one steam generator, even if both SGs are "dried out", until shutdown cooling or long term-post PCS break-core cooling is sufficient for decay heat removal."
>
> "It should also be emphasized that in all new EOPs, confirmation of adequate core heat removal requires the confirmatory indication of at least in-core temperature. In the Functional Recovery Procedure, if in-containment indications are considered lost, the continuing actions for PCS Heat Removal and Core Heat Removal include maximizing feed flow to available steam generators."
>
> "The new EOPs are structured such that, if in-containment vital instrumentation is considered lost then the Functional Recovery Procedure is implemented. If a decision is made to not rely on any in-containment instrumentation, the result is that (1) the PORVs are opened and all available safety injection (SI) pumps started and SI valves open; (2) auxiliary feed flow is maximized and directed to at least one S/G; (3) containment air coolers are placed in emergency configuration and containment spray manually maximized; and (4) all available service water and component cooling water pumps operated."

In developing a strategy to recover from this event, the licensee made the assumption that all in-containment instrumentation would be lost (Palisades Plant Emergency Operating Basis Document, EOP 6.0 page 38 of 41, EOP 9.0 page 25 of 133 and PRA, Appendix 6, page 2). This would require that the strategy explained above be performed.

Initially, the operators would complete EOP 1.0, "Standard Post Trip Actions." Upon completion of these actions, the operator would either initially enter EOP 6.0, "Excessive Steam Demand Event," or go directly to EOP 9.0, "Functional Recovery Procedure," if the operator determines that more than one event is in progress.

Upon entering EOP 9.0, the operator is directed to Success Path HR-3, "S/G heat sink and safety injection pump operation," based on actual S/G level being less than - 84% of wide range steam generator levels. The following actions are required by HR-3:

1) When PCS pressure falls below 1300 psia, trip all PCPs. [PCS pressure will decrease to below 1300 psia due to the rapid cooldown of the PCS caused by the steam break. The functional

recovery procedure, not assuming any specific single event is in progress, has the PCPs tripped in the event a LOCA could be occurring];

2) Operate all safety injection and charging pumps;

3) Isolate the MOST affected steam generator, [this requires the closing of both MSIVs which cannot be accomplished];

4) Verify feedwater criteria by either steam generator level [steam generator level is one of the instruments assumed lost by this event], or feedwater flow, in conjunction with steam generator levels [not available];

5) Check for acceptance criteria of HR-3 by SG level [not available], core exit thermocouples or loop cold leg temperature, [not available], and adequate safety injection flow; and

6) If HR-3 acceptance criteria is not met, then go to HR-4, "Once-Through-Cooling." [By using instruments inside containment which are not reliable, the operator may not be implementing once-through-cooling at this point in the procedure.]

Decisions made in using the EOPs as written rely upon instrumentation that has already been assumed to be inoperable and/or unreliable. The PRA agrees that in-containment instrumentation will be lost along with in-containment electrical components. "Violation of EEQ is assumed to lead to failure of instrumentation required by the operator to monitor systems and equipment within containment important to maintaining adequate core cooling." (PRA Appendix 6, Page 2)

The licensee continues to take credit for the in-containment instrumentation and electrical components when using EOP 6.0, "Excessive Steam Demand Event," and EOP 9.0, "Functional Recovery Procedure," for recovery.

The licensee maintains the PORVs closed with their block valves isolated. There is no guarantee that the block valves would be able to be opened in the harsh environment inside containment. "Currently to comply with 10 CFR 50, Appendix R, PORV breakers 52-196 and 52-224 are normally left open. This is to prevent the possibility of "hot shorts" simultaneously in control circuits for each PORV and its associated block valve, resulting in a LOCA." (Supplement 1 to NUREG-0737-Response to Draft Safety Evaluation - Procedures Generation Package, Attachment 1, page 3).

The licensee has failed to meet their commitment with regard to the following: (1) go to once-through-cooling; (2) ensure maximum feed flow to at least one steam generator; (3) maximize containment spray flow and place containment coolers in emergency alignment; and (4) maximize service water and component cooling water flow. Due to the facility relying on inoperable or unreliable instrumentation, there is no guarantee that the steps committed to would be performed. With the potential loss of instrumentation and other components upon diagnosis of the dual steam generator blowdown with failed open MSIV and

upon exiting EOP 1.0, the operator should immediately go to once through cooling to ensure the core remains cooled, then make attempts to regain SGs as a viable heat sink. This is considered a deviation from the specified commitment (255/93010-02(DRS)).

2.4.6.2 Loss of Instrumentation

The NRC concern, documented in the SER issued in 1986, was as follows:

> "Because some instrumentation inside containment may be affected by the two steam generator blowdown environment, operator training/procedures to cope with possible loss of information or misinformation is needed."

The licensee's response to this concern, documented in a letter to the NRC dated April 28, 1986, was as follows:

> "Caution notes will be added to the beginning of operator actions and Safety Function Status checks of the optimal recovery procedures for LOCA and Excess Steam Demand Event (ESDE-the events in which degraded containment atmospheric conditions are expected). These notes will state that due to the potential adverse effects of high containment temperature/pressure on in-containment instrumentation the operator should not rely on any single instrument, but rather observe confirmatory indications; and significance should be placed heavily on trending and less significance on specific instrument readings.
>
> "In light of these notes, the existing guidance in the LOCA and ESDE draft procedures for maintaining SG feed flow would prevent deliberately stopping auxiliary feedwater flow. Additionally, the SI pump throttling criteria, which require a high operator confidence level in primary temperature and pressure indications, would correctly prevent any operator action limiting SI flow.
>
> "The acceptance criteria for containment atmospheric conditions in the LOCA and ESDE procedures will be changed to have an upper limit of design pressure and temperature. Therefore, these conditions in safety function status check should "flag" operators into using the Functional Recovery Procedure".

The licensee, in stating that a caution note would be added to the EOPs, took credit for an existing caution. "During degraded Containment conditions, the operator should not rely on any single instrument indication due to large instrument errors. Alternate/additional instrumentation should be used to confirm trending of PCS conditions." This caution does not give the operator any guidance as to what instrumentation is lost or what alternate instrumentation is available. In addition, the basis documents for EOP 6.0 and EOP 9.0 state that all containment instrumentation is assumed lost for the MSLB with single MSIV failure scenario (EOP 6.0 Basis, Revision 4, page 38 of 41 and EOP 9.0 Basis, Revision 5, page 25 of 133).

The licensee met their commitment in this case, but failed to fully address the concerns expressed by the NRC. In particular, adequate procedure actions to cope with the loss of information or misinformation caused by the loss of instrumentation were not established. The inspectors reviewed the training records related to this event and found no records of training on loss of instrumentation. Some guidance is given in EOP 6.0 for narrow range steam generator level and pressurizer level, but this would only be applicable for a single uncomplicated steam line break (no other events in progress concurrent with the steam line break).

Operations personnel stated that they did not know what instrumentation would be affected, and that they would have to wait for the engineering staff to evaluate containment conditions and determine which instrumentation would be available for use. This evaluation would take place during the event.

The licensee failed to fully address the concerns raised by the NRC in response to item 2. Specifically, adequate changes to the procedures and operator training on loss of instrumentation related to this event were not developed. This is considered an open item (255/93010-03(DRS)).

2.4.7 Additional Areas Examined

2.4.7.1 Emergency Operating Procedures

The caution prior to step 12 in EOP 9.0, Success Path 9.0, states "If maintaining an SG heat sink is immediately needed to protect the Public Health/Safety and both SGs are required to be isolated, then the Shift Supervisor may direct departing from the SG isolation steps (reference 10CFR50.54X)." The basis for this caution states "If both SGs are affected, isolating both SGs will result in a violation of HR-3 safety function acceptance criteria. Only HR-4, "Once-Through-Cooling," would be available". This caution should direct the operators to the proper procedure, HR-4. The caution, stating that the Shift Supervisor may deviate from the procedure via 10 CFR 50.54x, provides no useful information. In this case, actions that can provide adequate protection are not specified even though they are available. In addition, 10 CFR 50.54x is invoked in an emergency when reasonable actions that depart from a license condition or a technical specification are needed to protect the public health and safety and no actions consistent with license conditions and technical specifications that can provide adequate or equivalent protection are immediately apparent. In this case, appropriate actions can be taken through HR-4.

EOP 6.0, Attachment 2, contains graphs to account for errors in pressurizer and SG narrow range level. This attachment in not referred to in the body of the procedure. It is referred to in footnotes in the Safety Status, but no guidance is given as to when these graphs are required. This was brought to the licensee's attention with no immediate response. This is considered an open item (255/93010-04(DRS)).

2.4.7.2 Training

The inspectors reviewed the training for this event and determined that the only training provided was on EOP 6.0. This training primarily consisted of being able to identify the "Most Affected" and "Least Affected" steam generators. It did not cover the background to this event or the commitments made to the NRC as to what actions were necessary to recover from this event. No training was conducted on the potential loss of instrumentation or actions needed to be taken in response to the loss of instrumentation. No training was conducted on EOP 9.0, which is the facility identified primary procedure for recovering from this event.

During the 1989 Emergency Operating Procedure Team Inspection, it was identified that, "there were no lesson plans for individual success path procedures of EOP 9.0, either in the simulator or the classroom phases of instruction." In the current training material, the lesson plan for EOP 9.0 covered the generic steps for EOP 9.0 and addressed the basis for each success path in an overview format. Individual training for each success path was not performed.

In discussions with the training staff, the inspectors noted that during the previous training cycle, a scenario had been run to cover this event. The following items were noted in a review of the Simulator Exercise Guide.

- Plant trip was initiated by the inadvertent closure of the MSIVs with failure of one MSIV to fully close.

- The training staff inserted leaks on both steam generators to simulate a MSLB in containment concurrent with failure of one MSIV. This was done in conjunction with operator over rides on the MSIV indications to give the appearance of the failed open MSIV. The dynamics of this event differed from those obtained by running a dual steam generator blowdown with one MSIV failed open.

- The criteria for ending this scenario was, at a minimum, for the operators to enter EOP 9.0 and determine which success path to use. The scenario could be allowed to continue until the most affected SG is isolated or until the crew had entered HR-4, "Once-Through-Cooling".

- No training was conducted on the potential loss of instrumentation inside containment.

2.4.7.3 Observations of Simulator Exercise

The following parameters did not appear to be properly modeled on the Palisades Plant specific simulator.

Containment Pressure - During the first scenario with all ECCS equipment available, the containment pressure peaked at 12 psig. During the second scenario with a loss of off site power and only one train of ECCS equipment available, maximum containment pressure was 22 psig.

Previous analyses indicated that these pressures are lower than what would be present during an actual event.

Containment Temperature - The maximum containment temperature identified during either scenario was 150°F. Previous analyses indicated that this temperature is lower than what would be present during an actual event.

As a result of the containment temperature and pressure discrepancies, it is believed that the training on this event had a negative effect. The operators were not presented with realistic values to enable them to make the required decisions. This tends to lead to a false sense of security and reliance on procedures that do not appear adequate to recover from this event. This is considered an open item (255/93010-05(DRS)).

3.0 Open Items

Open items are matters which have been discussed with the licensee, which will be reviewed further by the inspectors, and which involve some action on the part of the NRC or licensee or both. Open items disclosed during the inspection are discussed in Sections 2.4.4, 2.4.6.2, 2.4.7.1, and 2.4.7.3 of this report.

4.0 Exit Meeting

The inspectors met with the licensee staff (denoted in Section 1) at the site on May 4, 1993, for an exit meeting to summarize the purpose, scope, and findings of the inspection. A verbal summary of the inspection findings was provided to the licensee at that time. The inspectors discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspectors during the inspection. The licensee did not identify any such documents or processes as proprietary.