Florida Power
CORPORATION

June 30, 1988
3F0688-20

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

Attention: S. A. Varga

Subject: Crystal River Unit 3
Docket No. 50-302
Operating License DPR-72
Inspection Report 87-22

Dear Mr. Varga:

Florida Power Corporation provides our formal response to the Operational Safety Team Inspection (OSTI) observations included in Inspection Report 87-22. FPC apologizes for the delay in generation of the formal response. As discussed earlier, the items were tracked by internal systems even before receipt of the report and our corrective actions have not been delayed, only the documentation to the staff. Nevertheless, we will endeavor to avoid such delays in the future. Many of the items have been discussed extensively with the staff in other forums. If renewed dialogue is needed to assure mutual understanding of our responses, we will be pleased to support it.

Should there be any questions, please contact this office.

Very truly yours,

Rolf C. Widell
Director, Nuclear Operations Site Support

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FLORIDA POWER CORPORATION
RESPONSE
INSPECTION REPORT 87-22

OBSERVATION 87-22-01

Control of Design Basis Document Input as addressed in report section 4.1.1.

FPC Response

Florida Power agrees with the inspection team's concern in that unverified calculations are listed in the "Source" column of the Design Basis Document (DBD) without any clarifying note. Upon investigation into this concern, it was determined that the inconsistency is limited to the Decay Heat Closed Cycle Cooling (DC) Section of the DBD. This is because the DC section was developed as part of the DBD Pilot Program and the concept of creating unverified calculations as a "source" document was not utilized in further development of the DBD. Unfortunately, the initial DC section was not made to conform to the later philosophy.

To correct this inconsistency and assure an adequate understanding of the information in the DBD, FPC issued a temporary change to Section 6/6 of the Design Basis Document on March 29, 1988. A note was added, and accompanies the unverified calculations listed in the "source" column, which clarifies the intent of these calculations as information only.

OBSERVATION 87-22-02

Maximum Ultimate Heat Sink Temperature as addressed in report section 4.2.1.1.

FPC Response

This item has been discussed in detail through other correspondence and is not included in this response.

OBSERVATION 87-22-03

Maximum temperature of Decay Heat Closed Cycle Cooling Water during emergency operation as addressed in report section 4.2.1.2.

FPC Response

This item has been discussed in detail through other correspondence and is not included in this response.

OBSERVATION 87-22-04

Safety Classifications of Travelling Screen CWTS-2 as addressed in report section 4.2.1.3.

FPC Response

Florida Power agrees with the inspection team's conclusion that FPC had inconsistent documentation related to the safety classification of the Traveling Screen, CWTS-2, and that a technical basis for a non-safety related classification was not adequately presented during the inspection. As determined during the inspection, the current Safety Listing did classify CWTS-2 as non-safety related. This classification was confirmed verbally during the inspection by the original A/E, and subsequently, Training Lesson Plans ANO-83, ENG-133, and ENG-135 were revised accordingly. As a follow-up to the inspection, the A/E was requested to provide a written basis for the non-safety related classification. This justification was provided as a Safety Classification Review (Attachment 1) in accordance with Nuclear Operations Engineering Procedure SREP-1.

OBSERVATION 87-22-05

Cooling Water Flow to Safety-Related Components as addressed in report section 4.2.2.

FPC Response

PT-136, DC and SW System Flow Measurements and EGDG-1A KW loading due to ES pump, was written to address this concern. This item is being followed by Inspector Follow-up Item 88-05-02, and has been discussed in Inspection Report 88-05.

OBSERVATION 87-22-06

Diesel Generator Loading and Testing as addressed in report section 4.3.2.2.

FPC Response

This item has been discussed in detail through other correspondence and is not included in this response.

OBSERVATION 87-22-07

RCS Loose Parts Monitor Alarms as addressed in report Section 5.1.5.3.

FPC Response

Apparently FPC did not provide the inspection team sufficient evidence during their inspection to alleviate the teams concern that an inadequate evaluation was performed on the RCS Loose Parts Monitor System event of August 27, 1987. Considerable troubleshooting effort went into determining the most probable source of the noise/impacts and their effects on the plant.

At the time of the team's visit, FPC had just initiated the troubleshooting effort. Subsequent to the visit, a brief history of the event and FPC's resolutions have been documented and is available on site. FPC's conclusions are two-fold.

(1) The most probable source of the noise was a cycling internal vent valve.

(2) Such cycling did not represent a significant damage potential for short time periods.

Two partial RCS refills with cooler water temporarily decreased, or even stopped, the noise. Feeding the steam generator with a higher temperature feedwater increased it. These operations helped to establish and confirm the above conclusion. All possible troubleshooting actions available onsite were taken short of opening the reactor vessel.

Filling operations heading toward startup eliminated the noise as expected, thus supporting the cycling internal vent valve theory.

OBSERVATION 87-22-08

Purge and Vent Valve Seating, as addressed in report Section 5.2.1.

FPC Response

Florida Power Corporation agrees that the testing of the reactor building purge and vent valves as required is a personnel safety concern that needs further evaluation. A study is presently underway and is scheduled to be completed by October 1988 to determine a long-term solution concerning the maintenance of RB atmospheric conditions during various operating modes.

OBSERVATION 87-22-09

Control of Scaffolding, as addressed in report Section 5.2.3.

FPC Response

FPC understands the teams concern regarding the use of scaffolding near safety-related components. FPC is currently reviewing controls that may need to be addressed in FPC procedures.

OBSERVATION 87-22-10

Control of as-figured Information on Drawings, as addressed in report section 6.2

FPC Response

FPC has reviewed the NRC concern related to the timeliness of as-built drawings and FPC's selection of specific drawings that are as-built prior to turnover to Operations, after a modification is complete. The flow diagrams (302's) and electrical breaker drawings (201's), currently being as-built on an expedited bases, were selected jointly by operations and engineering personnel and were considered adequate to support plant operation pending the as-builting of the remaining plant drawings. As a result of this concern FPC has given priority to four additional series of drawings (205's - Instrument Loop Diagrams, 208's - Elementary Diagrams, 209's - External Wiring Diagrams and 210's - Internal Wiring Diagrams).

FPC recognized one factor affecting the overall efficiency of the drawing control process is the larger number of drawings maintained unnecessarily. A task force is being formed to evaluate the results of a study, of the FPC drawing system (60,000 plus drawings), which was previously prepared to review the as-builting process. This evaluation will determine which of the existing drawings need to be maintained or were only needed during construction and should be archived for historical purposes. The remaining drawings will be prioritized, based upon relative importance to the safe operation of CR3, for asbuilting purposes. Based upon the results of the review of the process and the total number of drawings found to be of highest priority, engineering procedures will be revised to reflect a requirement to issue as-built drawings within a given time frame, following the completion of the modification process.

A plan reflecting task force recommendations will be developed and implementation begun prior to 12/31/88.

Florida Power agrees with the identified concern regarding the lack of understanding for control of drawings as described in AI-405. As a result, maintenance senior shop supervisors were verbally instructed to use "working copies" while performing routine maintenance in the field. "Controlled copies" of drawings are to be used as a reference document in the shop area. This is an interim measure taken pending completion of the more comprehensive review of the FPC drawing system and control process described above.

OBSERVATION 87-22-11

Engineering Staffing, as addressed in report Section 6.3.

FPC Response

The NRC inspection team felt that the existing CR-3 engineering staff was too small to support the implementation of the on-going Configuration Management (CM) Plan and provide day-to-day technical support. They suggested that the previously prepared manpower study be revised to reflect this additional scope of work.

The magnitude of the engineering effort required to support the implementation of the CM Plan and its relative importance was recognized by FPC management and considered when the manpower study was prepared. FPC also realized that the existing CR-3 staff could not support this activity and still provide day-to-day support. Therefore, it was decided that a group, separate from the existing engineering functions, dedicated to the CM Program would be formed. This group, under the leadership of a separate project Manger with lead engineer positions (FPC personnel), would utilize contract engineering personnel and outside A/E support to meet the staffing needs of the effort. This group, including its Manager, three lead FPC engineers, and eight contract personnel, have been aggressively pursuing this activity. The current plan projects that in excess of 30 individuals (excluding A/E support) will be required throughout the effort, to support the plan implementation.

Once the peak workload effort is completed the manpower study does include sufficient FPC resources to maintain the CM Program and insure long lasting, high quality configuration control at CR-3.

OBSERVATION 87-22-12

Qualified Reviewer Process for PRC Reviews and Use of Subcommittees, as addressed in report Section 7.2.

FPC Response

In view of NRC's concerns regarding the lack of PRC procedures, FPC is considering the development of more formal guidelines for PRC activities. These proposed guidelines will contain information such as conduct of meetings, scheduling special presentations, method for submitting items for the agenda and other pertinent information. Meeting agenda and minutes is also being considered for inclusion in these guidelines. These changes will be complete by September 30, 1988.

In reference to the teams concern regarding use of Qualified Reviewers, the major responsibilities of the Qualified Reviewer have been included in administrative procedures. FPC has also discontinued the use of subcommittees.

OBSERVATION 87-22-13

Post Accident Sampling System Leakage, as addressed in report section 8.1.1

The items addressed in this observation represent several different areas and are addressed independently.

OBSERVATION (1) A:

General design criteria (GDC) 16, 60 and 64 were apparently not met for the post accident sampling system (PASS), resulting in unmonitored and uncontrolled radioactive gaseous releases. These releases were documented by NCORs 87-44, 87-48, 87-64 and 87-89. The leak-prone PASS piping lines are routed through the intermediate building, which has no radiation monitors and is vented to the atmosphere. Compression type mechanical fittings were used on the PASS samples lines instead of welded fitting, which is an apparent root cause of the leakage.

FPC Response:

General Design Criteria (GDC) 16, 60, and 64 as defined in 10CFR50 App A, Revised as of January 1, 1987 are not directly applicable to Crystal River Unit 3. As stated in section 1.4 of the FSAR, "Crystal River Unit 3 has been designed and constructed taking into consideration the proposed 10CFR50.34 Appendix A, "General Design Criteria for Nuclear Power Plant Construction Permits as published in the Federal Register (32FR10213) on July 11, 1967 which are applicable to this unit". FPC will initiate separate formal discussions with the staff to finalize the issue of Appendix A's GDC being substantially different than the CR-3 licensing basis. Informal discussions have not yet been conclusive.

OBSERVATION (1) B:

Although the licensee performed calculations which show no site radioactive release limits were exceeded for various uncontrolled releases, the team was concerned that similar system leakage under accident conditions when the PASS would be used have not been assessed.

FPC Response:

The intermittent operation of PASS with Maximum Hypothetical Accident sample concentrations and minor leakage should not exceed 10CFR Part 100 Limits. This is based upon the latest Gilbert Commonwealth MHA site dose calculations which did not exceed Part 100 quantities with leakage of a recirculation loop up to 4510 CC/HR and 50 gpm for 30 minutes occurring 24 hours after the accident as well as containment leakage. These calculations do not take credit for filtering of release, therefore a PASS leak into the Intermediate building would be bounded by this analysis.

OBSERVATION (1) C:

The plant staff could also receive unnecessary and excessive exposures or have their access to critical areas impeded. This aspect was not evaluated by the licensee.

FPC Response:

The vital area access report did not consider equipment failures (i.e. leakage), nor was this a requirement. However, PASS is designed with remote isolation and flushing capabilities. Thus if an equipment failure does occur the system can be isolated and dose rates reduced by flushing to allow the system to be repaired. This capability will protect plant staff and preserve access to critical areas.

OBSERVATION (1) D:

Long-term corrective actions are under evaluation by the licensee but no firm resolution or corrective action schedule had been established.

FPC Response:

The team has cited FPC's actions to date as an example of treating symptoms and not root causes. A better characterization is that FPC has taken immediate corrective action to deal with the unmonitored releases and simultaneously identified the existence of this release path as an item needing further study. This evaluation is part of REI 87-03-16-00. A preliminary study has been completed and a final evaluation should be completed by August, 1988. A definitive corrective action implementation schedule (if required) will be developed after the evaluation is complete.

OBSERVATION (1) E:

The team considered that the prior leakage and radioactive release episodes appeared to be reportable as required by 10 CFR 50.73(a)(2)(v)(c) as conditions that alone could have prevented fulfillment of the safety function of a system that is needed to control the release of radioactive material. This is because potential release paths are assessed in the FSAR and systems to control and monitor such paths, for example the auxiliary building ventilation system, were bypassed by this design.

FPC Response:

The FSAR assesses Miscellaneous Activity Releases from the Auxiliary Building (Section 11.2.3.2) and Radioactive Release from the Secondary Steam System (Section 11.2.3.3). A review of section 11.2.3.2 reveals that releases from the Auxiliary Building are to be discharged via the Auxiliary Building Ventilation System and is filtered (roughing, HEPA, charcoal) prior to release. The monitors in the vent measure and record the activity released and alarm if the setpoint is exceeded. However, the CR-3 Safety Listing pages 2-1 and 2-2 along with Flow Diagrams FD-302-752 and FD-302-695 indicate that Auxiliary Building Ventilation Exhaust is a non-safety related system. FSAR Section 11.2.3.3 discusses the potential gaseous and liquid paths for radioactive releases from the secondary steam system. A review of the CR-3 Safety Listing shows that section 11.2.3.3 describes components which are also considered as non-safety related. Therefore, we can conclude that leakage from PASS has not prevented fulfillment of the safety function of a system. PASS (and the other Reactor Building penetrations into the Intermediate building) are not discussed in the FSAR apparently because at the time of their design it was recognized that they were not in continuous operation and their impact was expected to be minor.

OBSERVATION (2):

The six toxic gas monitors had an extensive failure history dating back to 1982. During the last five years approximately one work request per month had been written for these monitors. Although no toxic gas releases were found to have occurred during the frequent periods when these monitors were out of service, their monitoring effectiveness was reduced. The licensee did not have a firm plan to correct the frequent failure problems, but replacement of these monitors was being considered.

FPC Response:

These and related issues are being discussed with the staff in separate extensive discussions. A brief summary follows:

- o The six present toxic gas monitors will be replaced by MAR 87-07-23-01. The new monitors will be reliable, low maintenance detectors that will have a shorter transport/detection time. The installation of these new monitors is tentatively scheduled for fall 1988.
- o Sulphur Dioxide (SO_2) detectors alarm in the CR-3 control room, and they also place the CR-3 Main Control Room HVAC in recirculation. These Interscan SO_2 monitors have proven to be reliable and accurate detectors. A surveillance procedure has been generated to ensure their operability.

- o A portable Sulphur Dioxide Interscan model toxic gas monitor has been installed in the CR-3 Main Control room for local air sampling. This monitor is also surveilled to ensure operability. The replacement of the present Toxic Gas System Monitors at CR-3 will alleviate failure problems with the present system.

OBSERVATION (3):

The licensee's inability to resolve the fatigue cracking problem should be evaluated further to determine adequacy of corrective actions.

FPC Response:

Florida Power Corporation (FPC) has reviewed all NCORs generated as a result of cracked welds associated with make-up pump vibration (see attachment 2). The modifications listed in each case has resolved the cracking problem. In one case (MUV-442 reported on NCOR 87-52) the piping had been shortened by MAR 79-07-07A. This MAR was written as a result of several NCORs written in 1979 and, in the case of MUV-442 provided a solution for the cracking problem for about 8 years until NCOR 87-52 identified a crack associated with this valve in 1987.

MAR 87-09-04-01 will be issued in February 1988 to replace all originally supplied drain valves with instrument valves (Lighter Weight).

OBSERVATION (4):

Long-term interface problems between the CR-3 nuclear staff and the adjacent FPC fossil unit illustrated past shortcomings of FPC's corrective action program for prompt problem resolution.

FPC Response:

Florida Power agrees that the technical interface between the CR-3 nuclear staff and the adjacent Fossil unit staffs was unsatisfactory and that adequate corrective action had not been taken to resolve deficiencies occurring during the time period 1984-1987. As pointed out in the inspection report, a task force consisting of fossil plant and CR-3 staff was formed in April 1987 to develop controls intended to prevent recurrence of the events described in the inspection report. In addition, a detailed study was accomplished between the Nuclear Operations Department and other FPC organizations including the Fossil Operations Department as well as organizations such as Substation Maintenance. The "Report of the Review of Interfaces Affecting CR-3" was approved for issue in September 1987. In November 1987, a "CR-3/Others Interface Matrix" was issued as a controlled document to all affected FPC organizations to highlight those interfaces which require the knowledge and participation of Nuclear Operations. Corresponding procedures and directives have been developed to ensure the satisfactory control of interfaces between the nuclear and non-nuclear staffs.

OBSERVATION 87-22-14

Reportability of Selected NCOR's as addressed in report section 8.1.2.

FPC Response:

Observation 8.1.2, together with item (1) of Observation 8.1.1, involves NR concerns with our reportability determinations. The following four events a presented as evidence of the NRC's concern.

1) Unmonitored and uncontrolled radioactive gaseous releases as noted in NCORs 87-44, 87-48, 87-64, and 87-89 (Observation 8.1.1, (1), pages 41-42).

2) Decay Heat Removal System piping hanger damage as noted in NCOR 86-33 (Observation 8.1.2, first bullet, page 43).

3) Failure to notify the NRC of tests impractical to perform as noted in NCOR 87-92 (Observation 8.1.2, record bullet, page 43).

4) Failure to document unacceptable steam generator pressure relief valve settings as noted in NCOR 87-42 (Observation 8.1.2, third bullet, pg. 43). Each of these events are addressed in attachment 3.

Based on the reevaluations of each NCOR identified in Attachment 3, FPC feels its present program adequate in determining reportability.

ATTACHMENT 1

Florida Power

SAFETY CLASSIFICATION REVIEW
Crystal River Unit 3

ITEM	COMPONENT
ALTERNATE RW SYSTEM	TRAVELING SCREEN - CWTS-2

Generic ☐ Yes ☒ No Application

SYSTEM	REQ. NO.	P. O. NO.	MMIS NO
RW & CW	N/A	N/A	N/A

Answer each question:

1. Does the item/service assure the integrity of the reactor coolant system boundary (i.e., "Pressure-retaining" as defined in ASME Boiler and Pressure Vessel Code)? Yes ☐ No ☒

Justification: (Answer the following questions - if both are Yes, check "Yes" above, otherwise check "No".

	Yes	No
• Is the item/service a part of reactor coolant system?	☐	☒
• Is the item/service pressure-retaining per Code?	☐	☒

Comments:

2. Does the item/service assure the capability to shut down the reactor and to maintain in a safe shut-down condition? Yes ☐ No ☒

Justification: Item automatically removes debris from entrance to RW System, which provides cooling water to essential plant systems. Automatic functioning is not a design requirement. See attached sheet for additional justification.

3. Does the item/service assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 10CFR100.11? Yes ☐ No ☒

Justification: Item automatically removes debris from entrance to RW System, which provides cooling water to essential plant systems. Automatic functioning is not a design requirement. See attached sheet for additional justification.

Is the answer to any question above "Yes"?

☐ Yes → Item/Service is Safety-related

☒ No → Item/Service is non Safety-related

NOTE:
If item is considered to be electrical equipment, the Environmental Qualification Requirements Review form must be completed to determine 10CFR50.49 applicability.

DESIGN ENGINEER	DATE	VERIFICATION ENGINEER	DATE	SUPERVISOR NUCLEAR ENGINEERING	DATE
[signature]	1/29/88	[signature]	1/29/88	[signature] for KEV	1/29/88

Rev. 2/87

RET: Life of Plant RESP: Nuclear Ops. Eng. 307 350

ATTACHMENT TO SAFETY CLASSIFICATION REVIEW
FOR TRAVELING SCREEN - CWTS-2

The Traveling Screen CWTS-2 is not classified as safety related because this component is not required to function to support safe shutdown of the plant using the Alternate Nuclear Service Seawater Cooling System (RW). The basis for this classification is as follows:

Flow rate of the RW System pumps is 23,800 gpm. At a minimum operating water level of 79 ft. elevation at the Unit 3 intake structure, approach velocity to the intake opening (at the face of the bar rack) is ~0.6 ft./sec. This velocity is low enough to minimize the potential for entrapment of fish and other objects, such as floating trash.

At the above noted flow rate, the velocity through a clean screen is 0.95 ft./sec. Traveling screens of this type are designed for velocities ranging up to 2.5 ft./sec.[1] through the screen. In this case, 2.5 ft./sec. corresponds to ~60% of the screen clogged. The time required to approach this level of clogging can range from several hours to several days. This elapsed time permits manual action to clean the bar rack and traveling screen, or at least to manually rotate the screen to expose clean screen elements. The bar rack will intercept a significant portion of approaching debris; thereby, reducing the potential for screen clogging.

Even if failure of the screen were to occur, the massed debris would enter the intake structure pre-chamber and float on the surface. At the minimum water level of 79 ft. elevation, at least 7-1/2 ft. of submergence is maintained above the top of the 48-inch intake pipe to the pump suction chamber. With a suction pipe entrance velocity of ~4.2 ft./sec. and 7-1/2 ft. of submergence, vortexing will not occur.[2] The debris will remain on the surface in the pre-chamber until shutdown of the alternate loop and cleaning of the chamber is feasible.

(1) Based on vendor data.

(2) Based on design experience and the relationship between available submergence and pipe inlet velocity for the design case.

Attachment (2)

HISTORY OF CRACK OCCURRENCE ASSOCIATED W/MUPs

NCOR	Valve #'(s)	Corrective MAR	Description of Correction
79-234	MUV-347	84-1-16-1	Replaced SW with BW
79-317	MUV-347	84-1-16-1	Replaced SW with BW
79-428	MUV-347	84-1-16-1	Replaced SW with BW
79-361	MUV-441 or MUV-442	79-07-07A	Shorten pipe nipples on MUP vents & drains.
79-395	MUV-291	84-1-16-1	Replaced SW with BW
81-475	MUV-71	83-11-2-1	Replaced piping w/flex hose
81-482	MUV-71	83-11-2-1	Replaced piping w/flex hose
81-483	MUV-71	83-11-2-1	Replaced piping w/flex hose
82-65	MUV-283	84-1-16-1	Replaced SW with BW
82-155	MUV-283	84-1-16-1	Replaced SW with BW
83-299	MUV-283	84-1-16-1	Replaced SW with BW
82-58	MUV-292	84-1-16-1	Replaced SW with BW
82-133	MUV-292	84-1-16-1	Replaced SW with BW
82-148	MUV-292	84-1-16-1	Replaced SW with BW
87-52	MUV-442	This re-occurrence of cracking about 8 years after NCOR 79-361 was resolved will be addressed by replacing the original drain valves with tubing and Instrument valves on MAR 87-09-04-01.	

Note: The valves listed above were specifically addressed by the listed NCORs but the valves in the same function & location on the other make-up pumps were also included in the listed MARs.

Attachment (3)

1) FPC's position on PASS design is presented in response to the observations grouped in 87-22-13. They provide the technical basis used in the reportability determination.

In summary, this series of events was considered "not reportable" based on the conclusion that the unmonitored release would not have been sufficient to conclude that the PASS design alone could have prevented fulfillment of the safety function of a system needed to control the release of radioactive material.

2) NCOR 86-33 - Decay heat removal system piping hanger damage occurred as noted in NCOR 86-33. This item was designated "not reportable" based on the cause of the problem being known and based on the knowledge that the known cause was already being reported. The cause of the event was the Decay Heat Removal Pump shaft break event of February 2, 1986, which was documented in NCOR 87-22 and reported to the NRC as LER 86-2. This LER discussed the significance of the damaged piping hangers and addressed the potential problems with Decay Heat Removal Pump operation during vortexing.

3) FPC does not consider 10CFR 50.55 a (g)(5)(iii) to be the basis for a report but rather is a condition for which relief can be sought. The failure to do so in a timely manner has been addressed and resolved in another forum.

4) NCOR 87-42 - Failure to document unacceptable steam generator pressure relief valve settings is noted in NCOR 87-42. The initial determination of this event was "not reportable", with a comment that "evaluation in progress may determine reportability to be required." Contrary to a statement in the NRC Observation, this event was in fact evaluated for reportability; however, the evaluations may have been too narrow in scope. On March 13, 1987, an Inter-Office Communication was written stating that the main steam safety valves (MSSVs) would have performed satisfactorily if required. In addition, an evaluation to demonstrate that the MSSVs did not put us outside the design basis for CR-3 was performed. These evaluations for reportability addressed the actual operability of the valves versus the design basis but ignored the apparent technical inoperability of the valves per the Technical Specification 4.7.1.1 requirement for the setpoints to be within 1% of the lift setting. Based on this TS setting requirement, all valves outside the 1% tolerance could be considered inoperable and the TS Action requirement of TS 3.7.1.1 would have to be complied with. The Actions of TS 3.7.1.1 require several actions, including restoration of the valve to operable status or reduction of the nuclear overpower trip setpoint per Table 3.7-1 within 4 hours; otherwise, be in at least hot standby within the next 6 hours and cold shutdown within the following 30 hours. If these actions were not complied with in every case where a MSSV was found out of tolerance, then the event would be LER reportable as a condition prohibited by the Plant Technical Specifications (10CFR50.73(a)(2)(i)(b).

FPC has reevaluated this NCOR for compliance with Technical Specification Action requirements. All Action requirements were met, therefore, NCOR 87-42 remains not reportable.