50-289

JAN 10 1986

MEMORANDUM FOR: Thomas E. Murley, Regional Administrator

THRU: Richard W. Starostecki, Director
Division of Reactor Projects

FROM: William F. Kane, Director
TMI-1 Restart Staff

SUBJECT: TMI-1 RESTART: NRC HOLD POINT FOR STARTUP AFTER COMPLETION OF RESTART TEST PROGRAM

The purpose of this memorandum is to summarize the completion of the special NRC inspection program for TMI-1 restart and to document my recommendation of January 3, 1986, that the licensee should be released to proceed beyond the sixth and final NRC hold point. Enclosure 1 is a summary of our assessment of the following areas: plant status and equipment operability, personnel knowledge and performance, testing program status, applicable license/restart conditions, allegations and enforcement actions, and the recent event at the Rancho Seco plant. Enclosure 2 provides the current status of all license/restart conditions generated by the steam generator and restart hearings.

During the period since the last hold point, the plant operated at steady-state 88% power conditions, the maximum achievable power due to steam generator water level operational limits. The operating environment continued to be similar to that of normal plant operation since there were only minor testing activities, except for the final turbine trip. Our coverage during this period was for twelve hours each day and covered all or part of the licensee's three operating shifts until January 2, 1986, at which time we shifted to 24-hour coverage to include the post-trip period up to steady-state power operation.

We concluded that the capabilities of licensee personnel were acceptable for sustained power operation. Based on our intensive review of licensed operators, we developed a high degree of confidence in their overall knowledge and performance skills. Although some inexperience was noted, non-licensed personnel implemented their duties reasonably well. There were no safety system challenges because of personnel error although a number of instances were identified in which the potential existed to cause such challenges.

Similarly, the material condition of the plant remained quite good. Licensee management had the plant ready for restart and sustained that material condition throughout the power ascension program with an aggressive preventive maintenance program. Licensee management continued to approach the testing program and power operation with caution and proceeded at a deliberate pace. Overall, the licensee's philosophy continues to be oriented toward nuclear safety, but there were signs at times during the past three months where that orientation was not aggressive. Recently, during preparation for release from the last hold point, the licensee's actions were oriented more toward getting the plant started

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than in conducting a complete review of problem areas. Although no unsafe conditions resulted, in the future, this aspect of the licensee's performance warrants close scrutiny by the specially assigned resident staff and by region-based inspectors.

Therefore, I recommend that you release the licensee from the final hold point and authorize a return to steady-state operation.

original signed by

William F. Kane, Director
TMI-1 Restart Staff

Enclosures: as stated

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ENCLOSURE 1

ASSESSMENT OF LICENSEE READINESS

TO PROCEED BEYOND 75% POWER LEVEL

1. Inspection Coverage

The TMI-1 Restart Staff continued its 12-hour shift coverage throughout the period, except during the New Year's holiday (4 hour coverage). At 7:00 a.m. on January 2, 1986, we commenced 24-hour coverage to monitor the planned trip from 88% power, post-trip activities, and return to power operation. Personnel from Region I and Battelle Northwest Laboratories manned the shifts. Inspection plans were developed for shift operational assessment, resident/region-based operational assessment, test witnessing and data evaluation, and supplemental inspections by region-based inspectors in specialty areas such as radiation protection. We completed the ninety-second consecutive day of our expanded shift coverage on January 3, 1986.

2. Testing Program

The power escalation test program culminated with the reactor trip on turbine trip test (TP 800/1) and subsequent emergency feedwater (EFW) pump auto-start test on loss of the reactor coolant pumps (TP 800/9). Additional tests which were performed at 88% power during this period included:

(1) TP 885/1, Turbine Generator Operations Test
(2) TP 800/5, Unit Load Steady-State
(3) TP 800/3, Thermal Expansion Check for Hanger Supports
(4) TP 849/1, ICS Tuning at Power
(5) TP 836/1, Feedwater System Operation and Tuning
(6) TP 846/1, Incore Thermocouple Operations Test
(7) RP 1550-08, Core Power Distribution
(8) RP 1550-05, Reactivity Coefficients
(9) SP 1303-1.2, Reactor Coolant Flow
(10) SP 1301-9.5, Reactivity Anomaly

We have reviewed the results of the above tests and conclude that the test data were acceptable and that test exceptions and deficiencies were properly documented and/or resolved. Overall, we concluded that the test and reactor physics procedures were implemented properly and data collection was in accordance with licensee procedures and regulatory requirements. In addition to the above tests, loose parts vibration monitoring baseline data continued to be taken as per procedure OP 1105-14 with no abnormal vibration or indication of loose parts detected. Further, the licensee restart program was effective in identifying equipment problems especially from the viewpoint of integrated system operations.

3. Plant Status and Equipment Operability

As of 7:00 p.m. on January 3, 1986, the plant was at hot shutdown conditions with the reactor coolant system temperature at 530 F and pressure at 2155 psig. The licensee had completed its evaluation of test data and equipment performance and was awaiting release from the NRC hold point.

No major safety-related equipment is out of service. Equipment is periodically taken out of service in accordance with the action statements of technical specifications to perform surveillance and preventive or corrective maintenance. Some events that occurred during the period which affected safety-related equipment are summarized below (details are described in the weekly status reports and applicable inspection reports for this period).

As part of the licensee's post-trip review, several equipment problems were noted. Reactor coolant system letdown isolation valve MU-V3 failed to shut after the reactor was tripped on January 2, 1986. The valve was tested successfully three times and the licensee was unable to determine the cause of the initial failure to close. The licensee decided to replace the time delay relay for the motor operator circuit and then tested the valve. The staff noted that the documentation of the test did not include a test of the reset function for the interlock that could possibly be affected by the time delay function. Subsequently, the licensee properly tested the reset function using an actual reactor trip signal. The licensee then declared the valve operable.

Further, during the trip, a portion of the station computer program failed such that the sequence of events printout data could not be recorded by the computer. This printout would normally be used to diagnose the cause of a trip. In this instance, the cause of the trip was known and the sequence of events data were not required to perform a post-trip review. The licensee attempted to find the specific cause of the problem but could not repeat the failure. The licensee now considers the sequence program operable and has committed to daily reviews to assure its continued operability.

Also, as a result of the turbine trip test, the licensee placed the steam supply for the turbine-driven emergency feedwater pump in a special test condition by maintaining a valve shut on one of the two supply lines while the valve in the other supply line was permitted to be opened during EFW actuation. During this test the safety valves on the steam supply line to the turbine did not actuate. Although the test procedure required only that all EFW pumps start on loss of all reactor coolant pump flow, the licensee noted that there may not have been sufficient steam pressure and volume to assure full flow from the steam-driven EFW pump (EFP-1) while supplying an operating

steam generator. The licensee declared EFP-1 inoperable. On January 3, 1986, the licensee readjusted the EFP-1 steam supply controller so that it would open the proper amount to supply a proper volume of steam at sufficient pressure. The pump was operated in a recirculation mode as per the surveillance test procedure and declared operable. The licensee plans to factor this test information into their long-term resolution to prevent actuation of EFP-1 steam supply safety valves during normal initiation of EFP-1.

Also, during the reactor trip, the indication from source range instruments NI-1 and NI-2 differed by about one to two decades. Trouble-shooting performed by the licensee indicated that the NI-1 count rate amplification gain was off the acceptance band. Correct gain adjustment was then applied. The licensee performed another surveillance test on the next day (January 3, 1986) and confirmed the previous day's adjustment.

During the reactor trip on January 2, 1986, the main steam safety valves again controlled steam pressure. Prior to the trip television cameras and personnel were stationed on the intermediate building roof to monitor which safety valve performance. The safety valves were again reseated by manually lowering the steam header setpoint after the third lift. The licensee will monitor the video tapes and perform a technical evaluation. The NRC TMI-1 Restart Staff concurs with the licensee evaluation that the relifting and reseating of the safety valves on a trip does not preclude continued safe operation of the plant. The TMI-1 Restart staff is not satisfied that the number of challenges to the safety valves has been minimized. A plan for corrective actions will be finalized by the end of the eddy current outage in April 1986.

A post-trip inspection of the reactor building by our shift inspectors revealed no major problems. Minor leaks were again noted by licensee representatives on several valves, including a steam generator instrument root valve and an EFW flange connection. The licensee's contractor applied a sealant (furmanite) to correct the leaks. Following the injection of furmanite into FW-V-1093, startup range level transmitter LT3 and operating range level transmitter LT5 for the "A" steam generator did not operate properly. The licensee concluded that the furmanite had plugged the valve. After the valve was operated and the line was pressurized, the blockage was flushed into the secondary side of the steam generator. The operating range transmitter tied to FW-V-1095 was returned to service. The startup range transmitter was left out of service for calibration since it has a redundant startup range for that steam generator. The licensee's documentation of the evaluation of the event and the initial furmanite repair was poor. The documentation did not consider explicitly the chemical and physical affects of the furmanite in the secondary system. Based on verbal discussions with the licensee and our own independent review and knowledge of the furmanite process, we conclude that the

furmanite will not have an adverse effect on plant operation. The furmanited valves are temporary repairs that the licensee has agreed to permanently repair during the steam generator eddy current testing outage currently scheduled for the spring of 1986.

In addition to the problems that occurred during the trip test, another problem was identified during operation at the 88% power plateau. The mechanical seal on a makeup pump failed and sprayed approximately 300 gallons of radioactive water into the makeup pump cubicle. The failure of the seal was attributed to a loose set screw that permitted the pump's thrust bearing internal components to loosen and shift, so that the pump's shaft moved slightly. The movement of the shaft caused excessive friction on the inboard mechanical seal and its resulting failure. The seal was replaced and the other two makeup pump shaft alignments were checked. The pump vendor recommends locking the set screw for the thrust bearing in place. The licensee plans to take the corrective action within two weeks of startup. The staff concurs with the licensee's plan.

4. Personnel Performance

Our intensive review of licensed operator personnel produced a good deal of information related to their knowledge and performance skills. The results of that review were very favorable. Our special interviews and discussions on shift confirmed a high level of knowledge of the facility design. Some evidence of weaknesses and inexperience in certain areas were identified; however, they were not unexpected and none indicated a generic weakness which would warrant immediate corrective action by the licensee. Our observation of performance revealed good communications and a good demonstration of their operational skills, especially when operating the relatively complex integrated control system in the manual mode. Operators implemented their duties well in response to unexpected events such as the unplanned reactor trip on December 1, 1985. The overall power escalation test program accomplished one of its main objectives, i.e., to provide actual operational experience for licensed operators. It was obvious that prior to criticality, licensee management provided the operators with extensive simulator time.

Similarly, we conducted an extensive review of nonlicensed personnel. In general, performance was acceptable; however, some areas if inexperience were noted, especially in the instrument and control area. Further, we saw a persistent problem with workers in the spaces having the potential to cause challenge to safety-related systems and/or operators. This group of personnel (as well as some licensed operators) were involved in the problem with the proper implementation of administrative controls for procedure implementation. Despite the licensee's strong program in this area, it was obvious that the program was not well understood at all working levels. The worker in the space issue and procedure control problems warrant continuing review by the specially assigned resident staff.

We believe that a strong training program produced the generally favorable results noted above. The power escalation complemented the licensee restart training efforts and it was a valuable training aid for all facility personnel.

5. Rancho Seco Loss of Power to Integrated Coolant System

On December 26, 1985, Rancho Seco experienced a loss of power to the integrated control system (ICS) which resulted in a reactor trip and subsequent reactor coolant system overcooling event. Other apparently significant operational problems occurred such as a loss of suction to a makeup pump. Information that we have received on the event is preliminary in that the NRC staff is just now getting into the analysis phase of the event review process.

We contacted representatives from AEOD and NRR to determine whether there were any known implications on the safe operation of TMI-1. Their response was that, although generic issues potentially could result from the NRC's investigative team review, no generic implications have been identified at this time by the staff that would preclude restart of TMI-1. The resident staff will continue to stay abreast of developments as the NRC's review of the Rancho Seco event continues.

6. Steam Generator/Restart Conditions

Three restart/license conditions became effective during this period. Enclosure 2 provides the updated inspection status for applicable steam generator and restart hearing-related conditions. All applicable conditions are tabulated to identify those that were reviewed by the staff and those that are open for future dated requirements (Enclosure 2).

7. Allegations

There is one open allegation in Region I for TMI-1 as noted in the previous hold point memorandum. This allegation does not preclude releasing the licensee from the final hold point.

8. Pending Enforcement Issues

Forthcoming Inspection Report 50-289/85-27 will describe violations of low severity level involving examples of failure to properly establish procedures, failure to follow or properly revise established procedures, and failure of the independent onsite safety review group (IOSRG) to comply with technical specifications requirements. These concerns were addressed, in part, in Inspection Report 50-289/85-26. Forthcoming Inspection Report 50-289/85-28 noted additional examples of the above problem but enforcement will not be taken pending acceptable licensee corrective actions.

9. Conclusions

The licensee's testing program essentially was completed in conformance with commitments and regulatory requirements resulting from the restart and steam generator hearings. The test program was successful in identifying residual component and overall integrated control system operational problems along with confirming expected results. The material condition of the plant remained quite good; and, although some deficiencies exist, none would affect safe power operation. The equipment problems identified in the licensee's post-trip review were adequately resolved or acceptable corrective action was planned subsequent to startup. The licensee could have been more complete in its review of the furmanite problem on the steam generator water level instrument valve. Overall knowledge level and performance of licensed and non-licensed operators remained at an acceptable level. Some performance problems persisted, such as the worker-in-the-spaces issue.

Licensee management philosophy continues to be oriented toward nuclear safety and worker radiation protection. However, licensee management should be more inquisitive in its review of plant equipment problems.

Overall, we conclude that the licensee's performance and plant material condition are acceptable to support continued power operations.

Enclosure 2

STATUS OF APPLICABLE LICENSE/RESTART CONDITIONS FOR

Steam Generator/Repair License Conditions

No.	Short Description	Inspection Status/History
2C(8)		
(1)	Prior to criticality, submit to NRC results of OTSG hot test program and management review.	IR-84-07 IR-85-16 IR-85-18 Complies
(2)	Confirm baseline leakage and shutdown if leakage exceeds 0.1 gpm above baseline (repair and retest).	IR-85-16 IR-85-18 IR-85-19 Complies
(3)	Complete post-critical testing on steam generator tubes	IR-85-30 (Issuance pending) Complete
(4)	Conduct eddy current testing per Table 3.3-1, NUREG 1019	Due 4/86
(5)	Quarterly report on long term corrosion level tests.	Not applicable, reports being sent to NRR
	(TMI-1 Restart Hearing Conditions)	
1		
(a)	Restriction of work in Fuel Handling Building (FHB) at TMI-1 during TMI-2 fuel handling in FHB.	IR-84-33 When required, inspection will be scheduled
(b)	Separate solid waste handling facilities.	IR-81-34, IR-82-03 IR-82-21, IR-84-33 (OI 84-33-01) IR-85-17 Complies
(c)	ESF filtration system for TMI-1 FHB (prior to TMI-1 fuel movements).	IR-84-33, 85-20 When required, inspection will be scheduled

No.	Short Description	Inspection Status/History
(d)	Liquid radwaste system isolation between TMI-1 and 2.	IR-82-21 IR-83-01 IR-84-33 IR-85-19 Complies
(e)-(k)	Unique shift manning requirements.	IR-85-19 Complies
(l)	Management system for feedback to operators.	IR-80-19, IR-81-33, IR-82-09, IR-82-16 IR-82-18, IR-83-01 IR-84-33 Complies
(m)	Restriction on former Startup and Test Director.	IR-82-19 IR-84-33 Complies
(n)	Preserve records for investigation of VV false certification	Not Applicable (Office of Investigations)
(o)	Operators to use instruments closest to saturation for HPI throttling until the Backup Instrument Readout (BIRO) system is safety grade.	IR-85-19 Complies
(p)	Until EFW is safety grade dispatch auxiliary operator to EF-V 30A/B.	IR-83-32 IR-84-33 IR-85-19 Complies
(q)	Reactor to be subcritical on hot standby with pressurizer heaters on emergency bus.	IR-83-32 IR-84-33 IR-85-19 Complies
(r)	Modify Emergency Plan for changing plant instrument capabilities.	IR-84-33 IR-85-17 Complies
(s)	Restriction on TMI-2 operating crew from operating TMI-1.	IR-85-19 Complies

(t)	Nuclear Safety and Compliance Committee (NSCC) to be retained.	IR-84-31 IR-85-25 Complies

No.	Short Description	Inspection Status/History
(u)	Restriction on individuals who replied to accident violations.	IR-85-19 Complies
2		
(a)	Before 5% power implement low power test program and training on natural circulation.	IR-85-18 IR-85-19 IR-85-22 IR-85-28 (Issuance pending) Complete
(b)	Before going above 48% power, initiate emergency feedwater on loss of main feedwater.	IR-85-25 Complete
(c)	Before 5% power, functional test 2-hour backup air supply to EFW	IR-85-22 Complies Complies
(d)	Before completion of the power escalation testing (PET), conduct a functional test of the reactor trip on loss of feedwater and turbine trip	IR-80-31, 81-03, 81-21 IR-81-14, 82-26, 84-33 IR-85-18 IR-85-25 IR-85-30 (Issuance pending) Complete
(e)	Before completion of PET, conduct functional test of inadequate core cooling instrumentation	IR-84-01 IR-84-33 IR-85-18 IR-85-30 (Issuance pending) Complete
(f)	Before completion of PET, conduct functional test of initiation of emergency feedwater pump on loss of reactor coolant pumps	IR-80-26, 81-28, 82-02 IR-82-26, 85-33 IR-85-18, 85-30 (Issuance pending) Complete

No.	Short Description	Inspection Status/History
3		
(a)	Prior to startup after cycle 6 refueling, upgrade EFW to safety grade	Estimated 4/87
(b)	Prior to startup after cycle 6 refueling, complete human factors deficiencies as noted in NUREG 0752, Supplement 1	IR-82-17 IR-83-09 IR-83-14 Complete
4	Within the first two years of restart, the GPUN qualification and requalification testing and training program shall be subject to indepth audit by independent auditor approved by Director, NRR	IR-83-19 IR-84-19 IR-84-25 IR-85-25 (Related inspections) NRR review
5		
(a)	October of each year submit progress report on installation of exact replica simulator	IR-85-24 Complies
(b)	Until replica simulator is available, provide one week training on Basic Principles Simulator.	IR-84-19 IR-85-25 Complies
(c)	Implement a leakage program for systems outside containment.	IR-83-14 IR-84-23 IR-85-18