U. S. NUCLEAR REGULATORY COMMISSION

REGION I

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Licensee:	GPU Nuclear Corporation
Facility:	Three Mile Island Station, Unit 1
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Inspectors:	Michele G. Evans, Senior Resident Inspector Samuel L. Hansell, Resident Inspector Douglas A. Dempsey, Reactor Engineer Mark Holbrook, INEL, NRC Contractor, Joseph Colaccino, Mechanical Engineer, NRR Dan Billings, Resident Inspector
Approved by:	Peter W. Eselgroth, Chief Reactor Projects Section No. 7

EXECUTIVE SUMMARY

Three Mile Island Nuclear Power Station
Report No. 50-289/96-05

This integrated inspection included aspects of licensee operations, engineering, maintenance, and plant support. The report covers a 7 week period of resident inspection; in addition, it includes the results of announced inspections in the areas of Inservice Surveillance Testing and Motor Operated Valve programs for unit 1.

Plant Operations

The operator's excellent response to the fuse clip failure for the station blackout diesel was an example of positive performance related to the application of good self checking techniques (Section M1.1).

Operations did not identify a degradation of Auxiliary Building and Fuel Handling Building Ventilation (ABFHV) system flow. In addition, the shift senior reactor operators did not document entry into the applicable TS Limiting Condition for Operation. The untimely ABFHV system operability determination was similar to an AB ventilation issue that was documented in NRC Inspection Report No. 50-289/96-02 (Section O1.2).

Operations performed and implemented multiple detailed on-line safety risk assessments for the decay heat valve modifications. The applicable Technical Specification limiting conditions for operation were entered and exited correctly for the safety related equipment outage times (Section M1.1).

Maintenance

The maintenance and surveillance test activities observed during this inspection were performed satisfactorily and demonstrated that the associated systems could perform their design safety functions. In particular, the station blackout (SBO) diesel air start solenoid valve replacement work activities were performed satisfactorily and should improve the SBO diesel reliability (Section M1.1).

Engineering

The inspectors were unable to close the TMI MOV program during this inspection. Adequate justification had not been provided to demonstrate the design-basis capability of a large number of your nontestable-valve groups (Section E1.1).

The design-basis capability of certain motor-operated valves (MOVs) was not demonstrated adequately for Generic Letter 89-10 program closure. For certain low-margin MOVs, the licensee did not adequately justify the valve factors (derived for nontestable valves) as the basis for demonstrating design-basis capability. Specifically, the approach used to apply certain industry and Electric Power Research Institute (EPRI) valve test data was informal,

not well controlled with respect to design input to safety-related calculations, and not technically sound (Sections E1.2 and E1.3).

The methods used to include load-sensitive behavior for rising stem MOVs were inadequate. The use of the MOVATS displacement measuring device (DMT) for determining load-sensitive behavior was inappropriate (Section E2.1).

The engineering modifications for decay heat valves DH-V-4A/B were very well planned, controlled, and implemented to address potential valve pressure locking issues (Section E2.1).

The TMI engineering and operations response to the Arkansas Nuclear One OTSG dryout and stuck open safety relief valve event was comprehensive, thorough, and displayed a strong initiative to address generic safety issues at other plants in order to reduce the potential for a similar impact at TMI (Section E4.1).

Plant Support

The radiography procedure changes and boundary controls significantly improved the radiological control personnels' ability to alert all plant workers about the conduct of radiography and ensure the exclusion boundaries were controlled properly. Procedure 6610-ADM-411.07 was revised satisfactorily to address the prior radiography problems (Section R8.1).

The Radiological Controls/Occupational Safety department customized self-checking training was a positive initiative to reinforce managements' expectations for each department (Section R5.1).

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Report Details

Summary of Plant Status

At the beginning of the period, Unit 1 was operating at 100% power. On June 29, 1996, the licensee reduced reactor power to 50% to repair main condenser tube leaks. The repair was completed satisfactorily and the unit returned to 100% power on June 30th. At the end of the period, the unit was operating at 100% power.

I. Operations

O1 Conduct of Operations (71707)[1]

O1.1 General Comments

Using Inspection Procedure 71707, "Plant Operations," the inspectors conducted frequent reviews of ongoing plant operations. In general, the conduct of operations was professional and safety-conscious; specific events and noteworthy observations are detailed in the sections below. In particular, the inspectors noted a repetitive concern related to the recognition and documentation of a Technical Specification (TS) operability determination that was documented in NRC Inspection Report No. 50-289/96-02.

O1.2 Auxiliary Building Ventilation Filter Operability Determination and Documentation

a. Inspection Scope (71707)

The Auxiliary Building and Fuel Handling Building Ventilation (ABFHBV) system is designed to maintain suitable and safe ambient conditions for operating equipment and personnel during normal plant operation. The system is also designed to minimize the release of radionuclides to the environment under postulated accident conditions by maintaining a negative building pressure. The required combined exhaust flow of the ABFHBV systems is 100,580 cubic feet per minute (cfm) to 130,691 cfm per TS.

On July 9, 1996, the inspector's noted in the Control Room log that the system flow had been logged by the night shift Reactor Operator at 100,560 cfm, which was below the TS minimum limit. The inspectors reviewed the ABFHBV system TSs, updated final safety analysis report (UFSAR), Control Room logs, and ABFHBV chart recorders. The inspectors also interviewed the system engineer and appropriate operations personnel to determine the actions taken in response to the low system flow.

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[1]Topical headings such as O1, M8, etc., are used in accordance with the NRC standardized reactor inspection report outline. Individual reports are not expected to address all outline topics.

b. Observations and Findings

The ABFHBV fans were shifted following the recognition of the low flow with the 'A' and 'C' fans secured and the'B' and 'D' fans placed in service. Subsequent readings were above the TS required minimum flow. A surveillance deficiency report (SDR) was written to document the ABFHBV low flow condition and the system was considered operable by operations. ABFHV system flow readings were recorded from the digital readouts in the control room at 9:00 a.m. and 9:00 p.m. each day to verify operability.

The system engineer presented a comprehensive plan to troubleshoot the ventilation system low flow condition. From July 9th through July 12th, troubleshooting continued with the ABFHBV system being shutdown for fan inspection, filter inspection, and filter change. On July 12th, at 4:30 p.m., the exhaust fans were restarted with no change in recorded flow. After operation management's review of the ABFHBV degraded flow, the SS entered the correct TS LCO due to the digital chart readings at the low end of the oscillation being less than the TS minimum limit of 100,580 cfm and the frequency of the oscillations indicating a degraded condition. The low end of the total flow oscillations were recorded as 99,400 cfm.

On July 14th, additional inspections of the dampers and ductwork revealed a broken manual damper. A temporary modification was written and the broken damper louvers were wired in the open position. System flow improved to approximately 110,000 cfm. A Plant Review Group meeting was held on July 15th to review the ABFHBV system condition and concurred with the operation management's decision to call the ABFHBV operable. As of the end of this inspection period the licensee continued to troubleshoot the system to identify additional causes of the degraded system flow.

The inspectors obtained copies of the control room ABFHBV charts showing a low flow condition on July 5th approximately 10,000 cfm below TS LCO minimum with no explanation documented on the chart. Operations personnel were questioned and the Operation's manager explained that the low flow was due to a 480 Volt electrical bus outage which caused a loss of control building fans. The same condition occurred again on July 12th when the same bus was de-energized to complete the breaker work. The low flow conditions existed for approximately one hour each. Operations personnel did not recognize the abnormal ventilation trends that resulted in the total ABFHBV flow that was less than the TS limit. In addition, the shift SROs did not document the applicable TS LCO when flow dropped below then recovered above the minimum value.

Operations management backdated the shift log on July 31st to include the correct dates and times for the three occasions when ABFHBV decreased below the minimum TS LCO value. The inspectors also obtained flow values from the daily logs showing a decreasing trend of ABFHBV flow over the previous two week period from 105,000 cfm to the low value of 100,560 cfm on July 9th. This flow was below the value of 118,000 cfm that was documented in October 1995 at the end of the last refueling outage.

c. Conclusions

Operations did not identify a degradation of Auxiliary Building and Fuel Handling Building Ventilation system flow. In addition, the shift senior reactor operators did not document entry into the applicable TS Limiting Condition for Operation. The untimely ABFHV system operability determination is a repeat problem that was similar to an AB ventilation issue documented in NRC Inspection Report No. 50-289/96-02.

II. Maintenance

M1 Conduct of Maintenance (62703, 62707, 61726)

M1.1 General Comments

a. Inspection Scope

The inspectors observed all or portions of the following maintenance and surveillance work activities:

- Job Order No. 106833, "'B' 125 VDC Ground Troubleshooting."
- Job Order No. 128218, "Station Blackout Diesel Air Start Valve Replacement."
- Job Order No. 121608, "'C' River Water Travelling Screen Troubleshooting Shear Pin Problem."
- Job Order No. 117416, "DH-V-4A/B Pressure Locking Modification."
- Job Order No. 122221, "BS-V-2B Limitorque Preventative Maintenance."
- Operations Procedure 1107-9, "Station Blackout Diesel Surveillance Test."
- Surveillance Procedure 1303-5.5, "Control Room Emergency Filter System Operability Test."
- Surveillance Procedure 1303-11.13, "Control Room Emergency Filter DOP and Halide Test."

b. Observations and Findings

Two positive observations were noted related to the station blackout diesel maintenance and associated post maintenance test run. First, the replacement of both air start solenoid valves with a new and improved design should result in fewer diesel slow starts and improved reliability. Secondly, control room operators (CROs) demonstrated an excellent questioning attitude during the performance of

Operations Procedure 1107-9, "Station Blackout Diesel Surveillance Test." The operator response to a fuse clip failure for the station blackout (SBO) diesel was a good example of positive performance related to the application of proper self checking techniques. The control room indications for the fuse clip failure were similar to a previous problem related to the 'A' emergency diesel. The improved procedure guidance that was incorporated since the previous problem aided the CROs in the early recognition of the SBO diesel trouble.

Positive observations were also noted related to the planned modification work on the decay heat injection valves DH-V-4A/B. The modification was performed to address motor operated valve pressure locking concerns. Multiple on-line maintenance risk assessments were completed for the Technical Specification (TS) components that were removed from service during the modification work. The applicable TSs were entered and exited correctly during the modification outage time. The work activities were well coordinated and executed by maintenance, engineering, operation, and radiological control personnel.

c. Conclusions

The maintenance and surveillance test activities observed during this inspection were performed satisfactorily and demonstrated that the associated systems could perform their design safety functions. The operator response to the fuse clip failure for the station blackout diesel was excellent and an example of positive performance related to the application of good self checking techniques. The engineering modification for decay heat valves DH-V-4A/B was very well planned, controlled, and implemented to address potential valve pressure locking issues. Plant operations performed and implemented multiple detailed on-line safety risk assessments for the decay heat valve modifications. The applicable Technical Specification limiting conditions for operation were entered and exited correctly for the safety related equipment outage times.

III. Engineering

E1 Conduct of Engineering (37551, 92903)

E1.1 Generic Letter 89-10 Motor-Operated Valve Program Review

Following this inspection, a management meeting was held between the NRC and the licensee on July 22, 1996, in the Region I office, to discuss the MOV inspection findings. The slides used by the licensee during this presentation are provided as an attachment to this inspection report.

a. Inspection Scope (TI 2515/109)

A sample of valves was selected for inspection that included examples of all methods utilized in the TMI's GL 89-10 program to demonstrate design-basis capability. The methods for demonstrating MOV design-basis capability included verification by: (1) valve-specific dynamic test at, or near, design-basis conditions,

(2) valve-specific test, linearly extrapolated to design-basis conditions, and (3) industry information provided by the Electric Power Research Institute's (EPRI) performance prediction program (PPP) and other nuclear facilities. The inspectors reviewed special test packages and engineering evaluations for the following selected MOVs:

DH-V-3	Decay heat system drop line isolation valve
DH-V-7A	Decay heat system pump discharge to makeup pump suction isolation valve
FW-V-92A	Startup feedwater block valve
FW-V-92B	Startup feedwater block valve
MU-V-36	Makeup pump minimum recirculation flow line isolation valve
MU-V-37	Makeup pump minimum recirculation flow line isolation valve
RC-V-2	PORV block valve
RR-V-5	Bypass valve for RR-V-6 pressure control valve

The GL 89-10 program scope consisted of 77 MOVs. A total of 36 MOVs were dynamically tested.

E1.2 Operator Sizing and Switch Setting Assumptions

a. Inspection Scope

The inspectors reviewed valve packages that established the thrust requirements for MOVs. The purpose of this review was to assess the licensee's justifications for assumptions used in MOV thrust calculations that form the basis for determining the design-basis requirements.

b. Observations and Findings

General Methodology

The thrust calculations typically utilized the standard industry equations. Mean seat diameter was used to calculate valve seat area. Valve factors were based on in-plant test results or from other industry sources, as specified by the licensee's grouping methodology. A stem friction coefficient of 0.20 was used for determination of actuator output thrust capability. A 2.9% bias margin and a 10.3% random margin were used to address load-sensitive behavior (also known as "rate of loading") for those rising stem MOVs that were not dynamically tested. A bias margin of 5% was included to account for potential future valve degradations.

Minimum thrust requirements for setting of actuator torque switches were adjusted to account for diagnostic equipment inaccuracy and torque switch repeatability.

Valve Factors - Valve Grouping

The MOVs were divided into valve groups based on valve manufacturer, valve type, and ANSI pressure class rating. In-plant test data was used, when available, for justification of valve factors for nondynamically-tested MOVs. If in-plant data was not available, then data from other utilities and the EPRI PPP were used to establish valve factors. However, some of the in-plant dynamic test data were not considered adequate to justify applied valve factors. This placed greater burden on obtaining adequate industry information to justify program assumptions.

The following are examples where grouping methods did not meet the intent of GL 89-10, Supplement 6, guidance for developing an adequate bases for program assumptions. The valves in these groups also had small apparent thrust margins.

Group #6

Valve Group #6 (FW-V-92A & 92B - close safety function) consisted of 6" Crane 900 # flex-wedge gate valves. The licensee had assigned a 0.42 valve factor for this group, which had a design-basis close differential pressure of 580 psid. The valve factor was based on a friction coefficient obtained from the ambient fluid temperature, pumped flow closing test of Valve #14 from EPRI's PPP. Two other valve factors from similar valves at two other PWRs were also considered. No specific analysis method was applied to the industry data, except that more weight was given to the EPRI test result. One utility had conducted two dynamic tests on a single MOV, and obtained valve factors of 0.363 (Test 9) and 0.463 (Test 10). The other utility's dynamic test obtained a valve factor of 0.365. The inspectors noted that the tests were conducted at differential pressures that were more than double the design-basis conditions specified for the TMI Group #6 MOVs. EPRI testing has shown that valve factors tend to decrease as differential pressure increases. Therefore, it is reasonable to expect that the industry valve factors would be higher if the dynamic tests had been conducted at conditions closer to 580 psid.

The inspectors noted that Valve Group #6 included an additional 12 data points from industry (outside TMI) dynamic tests of six other MOVs. However, these valve factors were excluded from consideration ("given less weight"), because the valve/actuators were orientated differently than FW-V-92A & 92B. No technical justification was provided to support these exclusions. A review of this data showed that every excluded data point, except for one, was higher than the applied 0.42 valve factor.

Group #9

Valve RC-V-2 (PORV Block Valve) in Valve Group #9 was a 2.5 "Velan 2500 # flex-wedge gate valve. The licensee had assigned a 0.40 valve factor for this group,

which had a design-basis close differential pressure of 2367 psid. This valve factor was based on a friction coefficient obtained from the ambient fluid temperature, high pumped flow closing test of Valve #13 from EPRI's PPP, and in-plant test results obtained from two 1500 # Velan gate valves (MU-V-36 & 37). The close valve factors for these two valves were less than 0.40. However, the inspectors noted that MU-V-36 & 37 were of a different pressure class than RC-V-2. Further, when selecting EPRI data, GPUN did not use the highest valve factor observed by EPRI prior to hard-seat contact (0.452) for Valve #13. Instead, a valve factor that EPRI identified as the valve factor present at flow isolation was used in the calculation. The use of flow isolation valve factors when applying test data to similar nondynamically-tested MOVs is not appropriate due to the valve specific nature of determining flow isolation.

The industry data acquired for Valve Group #9 were for 1500 # gate valves and did not include any valve factors for 2500 # gate valves similar to RC-V-2. A close valve factor of 0.424 was obtained from a similar 2.5" Velan 1500 # gate valve at ANO. However, this data was excluded ("given less weight") on the basis that the data came from another nuclear facility (instead of from in-plant testing or EPRI testing). Valve factors obtained from 20 other 1500 # Velan gate valves were excluded on the basis that they were slightly larger in size (3" to 4") than RC-V-2. The licensee stated that the average valve factor for this other industry data was 0.56.

In summary, the inspectors considered the valve factor justifications for Valve Groups #6 and #9 (RC-V-2 only) to be weak. The close thrust margins (based on the current valve factor assumptions) were as follows: FW-V-92A - 2.2%; FW-V-92B - 3%; and RC-V-2 - 20.1%. Appropriate technical justifications for the selection of the Group #6 and #9 valve factors are necessary to establish the valve's design-basis capability, prior to GL 89-10 program closure.

<u>Valve Factors - EPRI Data</u>

The licensee used disc friction coefficients obtained from EPRI's PPP to support valve factor justifications for several valve groups. The inspectors had several concerns with the application of these data.

Some valve groups (e.g., Valve Groups #6 and #13) used an individual EPRI disc friction coefficient as the primary basis for the selected valve factor. Other valve groups (e.g., Valve Group #14) listed additional industry data, but the selected valve factor for the group closely matched the EPRI disc friction coefficient. The EPRI disc friction coefficients were documented in EPRI report, "EPRI MOV PPP Update of Results and Specifications and Drawings for Flow Loop Test Valves," dated December 14, 1993. Consistent with the guidance provided in Supplement 6, individual data points do not demonstrate similarity of valve performance and, therefore, are not adequate justification for valve factors applied to nondynamically-tested MOVs.

The EPRI PPP determined apparent disc friction coefficients at several points during a valve stroke. These friction coefficients are not equivalent to valve factors and should be converted to valve factors by accounting for the angle of the valve's seating surface. These conversions were not made, which resulted in the use of nonconservative valve factors (for the closing direction).

EPRI PPP disc friction coefficients may not be reliable for use as an individual data point due to the lack of preconditioning (in some cases) and the general practice of removing apparent parasitic loads from the measured force requirements before calculating the apparent friction coefficient. Removal of parasitic loads results in a nonconservative thrust requirements if the disc friction coefficient is applied to a different valve and the parasitic loads are not added back into the minimum thrust requirement. The EPRI PPP does not identify when parasitic loads were present.

Valve Group #13 consisted of MS-V-2A and 2B (12" Walworth 600 # flex-wedge gate valves) that are main-steam system valves that isolate the line that provides steam to the auxiliary feedwater turbine. A valve factor of 0.47 was selected based on early PPP test results of EPRI Valve # 31, which was a 12" Walworth 150 # flex-wedge gate valve that was tested under ambient temperature, low pumped flow conditions. The inspectors questioned the selection of this EPRI valve based on the lower pressure class and the difference in the test conditions as compared to the design-basis conditions identified for Valve Group #13. The inspectors indicated that EPRI Valve #30 (6" Walworth 900 # flex-wedge gate valve) would have been a better match because of the typical similarity of construction between 600 # and 900 # gate valves, and because the EPRI testing for Valve #30 was done under high temperature fluid flow conditions. These test conditions may be more applicable should MS-V- 2A and 2B be called upon to close against a downstream steam line break.

The inspectors noted that Valve Group #9 used an EPRI disc friction coefficient (0.287 - EPRI Valve #13) that was measured at flow isolation. This same EPRI test had a maximum close friction coefficient of 0.452. Due to the arbitrary nature of determining flow isolation, the observed flow isolation friction coefficients are valve specific. As stated in the NRC's safety evaluation report (SER) by the Office of Nuclear Reactor Regulation of Electric Power Research Institute Topical Report TR-103237, "EPRI Motor-Operated Valve Performance Prediction Program," dated March 15, 1996, "EPRI states that the model output for flow isolation is 'theoretical' flow isolation position that is for information only and is not to be used to establish thrust requirements in accordance with the EPRI methodology."

The NRC staff reviewed the EPRI performance prediction model (PPM) software as documented in the above referenced SER. The staff's endorsement of the PPM (with the conditions stated in the SER) only covers the use of the PPM software. The SER does not accept use of the PPP's individual disc friction coefficients.

Because of the above concerns, the inspectors did not consider the use of EPRI's PPP friction coefficients to be appropriate for justification of valve factors used by

their grouping methodology. A sound technical justification for the use of EPRI PPP data in selecting valve factors is necessary prior to GL 89-10 program closure.

Valve Factors - Industry Data

The licensee obtained valve factors information from industry sources via phone conversations with other licensees. The inspectors noted a lack of thorough supporting documentation for the valve factors used from industry sources. Specifically, the licensee had not thoroughly documented the following information:

- The test conditions (i.e., pressures, flows, fluid temperatures) and system configurations should be documented. Pressure instrument locations and methods used to record differential pressure should be clearly understood. This is especially important for cases where test results appear to be abnormally low.

- How the force point on the diagnostic trace was selected (e.g., at flow isolation, hard seat contact, or highest force up to hard-seat contact).

- What was the valve seat diameter measurement used by the valve factor calculation and whether this diameter was obtained by measuring the valve's orifice diameter, the mean seat diameter, or by some other point of reference.

- The general method used to calculate the valve factor. This would include knowing how packing loads and stem rejection loads are determined and removed from the force measurements. It would also include knowing if parasitic loads were removed before calculating the valve factor.

This information is necessary to ensure that reliable data is used to establish design-basis requirements. Documentation for industry data, used to determine valve factors, is necessary prior to the closure of the GL 89-10 program.

Valve Factors - Nondynamically-Tested MOVs

The MOV program included several valve groups where the valve factor justifications were inadequate:

Group 1 - 3" & 4" Aloyco 150 # Split-Wedge Gate Valves - Valve Factor = 0.50

One MOV (BS-V-2B) was tested; however, the licensee was unable to determine a valve factor due to a lack of observable differential pressure effects in the diagnostic trace. One EPRI prototype test friction coefficient was considered. Twelve industry valve factors (from eight MOVs) were obtained over the telephone or via telefax communications. Some industry valve factors were larger than 0.50 and others were less than 0.50. Therefore, it was not clear how these data were

analyzed and TMI had not determined the disc orientation of the valves tested by the industry sources (see the next section addressing concerns related to disc orientation of Aloyco split-wedge gate valves).

<u>Group 2 - 14" Aloyco 150 # Double-Disc Gate Valves - Valve Factor = 0.60</u>

In-plant or industry data had not been collected to support this assumption.

<u>Group 3 - 4" Aloyco 300 # Split-Wedge Gate Valves - Valve Factor = 0.60</u>

One MOV (DH-V-7B) had been tested, and the licensee measured a valve factor of 0.66 for the open safety direction. This valve factor was not used (see Section 3.0 of this report). Four industry valve factors (from two MOVs) were calculated by TMI by reviewing the diagnostic traces. Five other industry valve factors (from three MOVs) were obtained over the telephone or via telefax communications. Some industry valve factors were larger than 0.60, and others were less than 0.60. Therefore, it was not clear how these data were analyzed to arrive at a 0.60 valve factor.

<u>Group 9 - 2" Velan 1500 # Flex-Wedge Gate Valves (MU-V-36/37-open only) - Valve Factor = 0.40</u>

In-plant open test data was not available for MU-V-36/37. One EPRI prototype open test friction coefficient was not used. Nine industry valve factors (from seven MOVs) were calculated by TMI by reviewing the diagnostic traces. Ten other industry valve factors were obtained over the telephone or via telefax communications. None of the industry data were used for the open safety direction.

<u>Group 10 - 4"/6"/8" Walworth 150 # Solid Wedge Gate Valves - Valve Factor = 0.40</u>

One MOV (IC-V-2) was tested and the licensee measured a valve factor of 0.44 for the open safety direction. This valve factor was not used (see Section 3.0 of this report). Three other MOVs were dynamically tested, but TMI was unable to determine a valve factor due to a lack of observable differential pressure effects in the diagnostic trace. One EPRI prototype test friction coefficient was considered. Six industry valve factors (from two MOVs) were calculated by TMI by reviewing the diagnostic traces. One industry valve factor was obtained over the telephone or via telefax communications. Some of these industry valve factors were larger than 0.40, and others were less than 0.40. Therefore, it was not clear how these data were analyzed to arrive at a 0.40 valve factor.

<u>Group 13 - 12" Walworth 600 # Solid Wedge Gate Valves - Valve Factor = 0.47</u>

No in-plant or industry data was available to support this assumption. One EPRI prototype test disc friction coefficient was the basis for the selected valve factor early in the program's development. However, EPRI increased this friction

coefficient in a later PPP test report, and TMI did not update their assumptions to account for this change.

Group 14 - 6" Walworth 900 # Flex-Wedge Gate Valve - Valve Factor = 0.40

No in-plant data was available to support this assumption. One EPRI prototype open test friction coefficient was considered. Nine other industry valve factors (from four MOVs) were obtained over the telephone or via telefax communications. Some industry valve factors were larger than 0.40, and others were less than 0.40. Therefore, it was not clear how these data were analyzed to arrive at a 0.40 valve factor.

Group 16 - 10"/12"/14" Walworth 1500 # Flex-Wedge Gate Valves - Valve Factor = 0.50

Three MOVs were dynamically tested; however, the licensee was unable to determine the valve factors due to a lack of observable differential pressure effects in the diagnostic trace. Industry valve factors for three MOVs were obtained over the telephone or via telefax communications. Some of these industry valve factors were larger than 0.50, and others were less than 0.50. Therefore, it was not clear how these data were analyzed to arrive at a 0.50 valve factor.

While the MOVs in the valve groups listed above appear to have adequate margin based on current assumptions, the basis for the thrust/torque requirements have not been well established. Additional information (e.g., results from EPRI's PPM or other applicable industry data) is necessary to support current thrust/torque requirements for these MOVs prior to GL 89-10 program closure.

Valve Orientation

Industry experience has shown that Aloyco split-wedge gate valve performance is sensitive to valve disc orientation. The licensee obtained Aloyco valve factor data from other industry sources for use in their grouping methodology. However, the test valves' disc orientations were not provided by the industry sources. Further, the engineering staff were not sure of the orientation of certain Aloyco split-wedge gate valves. Therefore, some of the industry data may not be applicable to the valves at TMI. Further evaluation is necessary to obtain orientation information for the industry data that was acquired, remove nonconservative data points, and reassess the assigned valve factors for Valve Groups #1 & #3.

Load-Sensitive Behavior

The load-sensitive behavior data was provided in Appendix J of TMI's program description. The average of the data for rising stem MOVs was 2.29%, with two standard deviations of 10.29%. The average value was increased to 2.9% and was added to the thrust requirements as a bias error. The two standard deviation value (10.3%) was combined with other errors and uncertainties in a square root sum-of-the-squares methodology.

The MOVATS displacement measuring device (DMT) was used during dynamic testing to determine thrust output. The DMT was calibrated to indicate thrust by correlating the spring pack displacement to a MOVATS stem strain ring's (SSR) thrust measurement during a static test. However, the MOVATS DMT will not detect the presence of load-sensitive behavior that may occur during a dynamic test. This is because load-sensitive behavior is the result of an increase in stem friction coefficient caused by the higher stem load present during a dynamic test. If the stem friction coefficient increases during the dynamic test from what was present during calibration of the DMT, the DMT measurement will overestimate the actual thrust that was applied to the stem. Therefore, the DMT provides unreliable thrust values during dynamic testing, an is unable to detect the presence of load-sensitive behavior. Appendix J analyzed data from 23 rising stem MOVs, of which 21 were tested using the MOVATS DMT. Based on the inability of the DMT to identify the presence of load-sensitive behavior, only two of the data points (which were measured using the VOTES diagnostic system) were valid. The inspectors did not consider two data points adequate to justify the licensee's load-sensitive behavior assumptions for rising stem MOVs and were unable to close this aspect of TMI's generic letter program.

TMI used the VOTES diagnostic system to diagnostically test rising rotating-stem globe valves. Because the VOTES system directly measures the thrust applied to the valve stem, it can detect load-sensitive behavior. TMI tested eight rising rotating-stem globe valves. The VOTES equipment was used for four of these dynamic tests and the MOVATS DMT was used for the remaining four tests. Therefore, the four VOTES tests provided valid load-sensitive behavior results. Appendix J identified the average of the rising rotating-stem globe valve data as 6.22%, with two standard deviations of 17.74%. The inspectors independently analyzed the four VOTES data points and obtained slightly higher values. However, given the uncertainty involved with analysis of this small number of data, the inspectors considered the licensee's assumptions for rising rotating-stem globe valves to be adequate. The licensee will need to collect additional load-sensitive behavior data, to improve their justification, as part of their periodic verification program.

<u>Stem Friction Coefficient</u>

The thrust calculations used a 0.20 stem friction coefficient assumption. Test results were analyzed in a draft document that was generated to support TMI's MOV trending program. The inspectors noted that one gate valve and two globe valves had stem friction coefficients significantly above 0.20. All remaining test data points were ≤0.20. Therefore, the inspectors determined that the licensee's basis for the use of a 0.20 stem friction coefficient assumption was adequate. The licensee will need to increase their confidence level in this assumption by obtaining additional stem friction coefficient data in the future, as part of their MOV periodic verification program.

Diagnostic Equipment Uncertainties

On September 21, 1993, GPU Nuclear Corp. submitted a response to the NRC's Generic Letter 89-10, Supplement 5, "Inaccuracy of Motor-Operated Valve Diagnostic Equipment." This submittal documented TMI's actions taken to resolve issues related to the open calibration of the MOVATS DMT using a load cell. To resolve this issue, 28 MOVs were reviewed and found to have acceptable settings. TMI also uses the VOTES diagnostic system. However, all test data was acquired with a version of the VOTES software that included torque correction factors. Therefore, no actions were required for existing VOTES tests.

During discussions with TMI personnel, the inspectors identified that the licensee's MOV switch setting methodology did not account for the diagnostic equipment uncertainty associated with use of the DMT to determine actuator torque output. Accounting for these uncertainties is important to ensure that torque limits are not exceeded. The licensee will need to conduct a screening to determine the impact of this error, take action as necessary, and to correct diagnostic procedures prior to MOV program closure.

c. Conclusions

In general, TMI's did not implement a rigorous analysis method for application of industry information when determining valve factors for nondynamically-tested MOVs. Given the many concerns, the inspectors did not find TMI's methods for justifying the valve factors and load-sensitive behavior assumptions used in design-basis thrust equations to be adequate for GL 89-10 program closure.

10 CFR 50, Appendix B, Criterion III, and the GPU Nuclear Operational quality Assurance Plan require that design control measures be established for verifying and checking design inputs. The failure to establish proper design control measures, when determining the valve factors for certain safety-related valves, is a violation of these requirements **(50-289/96-05-01)**.

E1.3 Design-Basis Capability

a. Inspection Scope

The inspectors reviewed TMI's "Program Description for NRC Generic Letter 89-10 Motor-Operated Valve Program" valve test packages and associated test reports for the selected MOVs. The purpose of this review was to assess TMI's efforts to establish design-basis capability for all MOVs in their GL 89-10 program.

b. Observations and Findings

Thrust Calculations

The dynamic test of DH-V-3 measured a close valve factor of 0.56 and an open valve factor of 0.64. The thrust calculation for D H-V-3 used an inappropriate valve

factor of 0.50. The licensee revised the thrust calculation using the measured valve factors. The revised calculation indicated that the current settings were adequate because the design-basis differential pressure was being reduced to remove a mispositioning scenario. The inspectors found other cases where measured open valve factors were not applied (i.e., DH-V-7B and IC-V-2). TMI indicated that these valve factors were not valid and should not be applied, because: (1) DH-V-7B; differential pressure effects were not detected and unwedging loads were used to calculate the valve factor, and (2) IC-V-2; differential pressure effects were not detected and the force equivalent to spring pack preload was used to calculate the valve factor. The inspectors noted that TMI's grouping justifications did not contain this information. The licensee will need to revise the appropriate program documents and thrust calculations to incorporate tested valve factors prior to program closure.

Torque Switch Repeatability

The TMI Program Description, Data Sheet 2, Appendix H, was used to document the post-test acceptance criteria for static and dynamic testing. Completion of this checklist was required prior to returning an MOV to service. The inspectors noted that the dynamic test margin assessment did not consider the added margin required to account for torque switch repeatability. The licensee agreed with this observation and will revise the checklist to include this margin. Further, TMI will review the impact of this omission. These items will require resolution prior to MOV program closure.

Test Data Extrapolation

The dynamic test review checklist for the dynamic test of RR-V-5 (Bypass Valve for RR-V-6 Pressure Control Valve - 10" Pratt 150 # Butterfly Valve), noted that the dynamic test differential pressure was 128 psid, which was approximately 90% of the design-basis differential pressure of 141 psid. However, the dynamic test review checklist identified this as a full differential pressure test and did not extrapolate the hydrodynamic test results to design-basis conditions. TMI personnel were unable to document why test results were not properly extrapolated, but postulated that the extrapolation was not done, because the test differential pressure was unknown at the point of maximum load during the open stroke. While the lack of differential pressure information may hinder accurate determination of design-basis torque requirements, it does not justify the documented evaluation where the test was considered a full differential pressure test. Similar concerns were identified for valves NR-V-4A and NR-V-4B. The inspectors also noted that, if the measured seating or unseating torque bounded the measured hydrodynamic torque, TMI assumed that the seating/unseating torque was bounding and that no further extrapolation to design-basis conditions was needed. However, the hydrodynamic torque should be extrapolated before making a comparison to the measured seating/unseating torque. The licensee will need to complete these extrapolations and reassess the adequacy of the vendor's (or TMI's internal) calculations.

Post-Maintenance Testing

The licensee's post-maintenance requirements are contained in Corrective Maintenance Procedure 1410-V-10, "Gate, Globe, and Needle Valve Maintenance," Rev. 29, dated June 6, 1996. The inspectors noted that this procedure did not clearly indicate when dynamic test conditions would be considered for a post-maintenance test requirement. Licensee personnel stated that they would clarify their procedures to indicate when dynamic testing would be considered.

c. Conclusions

Several deficiencies were identified with the MOV thrust calculations and test data differential pressure extrapolations. Other weaknesses were identified in the methods used to verify test margins for dynamically-tested valves and procedure for establishing post-maintenance test criteria. The inspectors concluded that additional attention is required to improve the current GL 89-10 program prior to program closure.

E1.4 GL 89-10 Program Scope

a. Inspection Scope

The inspectors reviewed the licensee's technical bases for removing the main steam isolation valves, turbine bypass line, and steam bypass to the condenser isolation valves, and the emergency feedwater turbine steam supply isolation valves from the GL 89-10 program.

b. Observations and Findings

Main Steam Isolation Valves (MS-V1A/B/C/D)

The Generic Letter 89-10 program description, Table 1, "Selection of Valves For GL 89-10 Program," states that the safety function of the MSIVs is only the check valve function. The licensee concluded that the motor operator does not perform a safety function. The licensee's MOV engineers stated that the MSIVs are not credited in UFSAR accident analyses for isolating a faulted steam generator due to the small steam generator secondary side water inventory, the magnitude of the blowdown, and the relatively slow valve stroke time.

The inspectors noted that the licensee's basis for exclusion of the MSIVs from the MOV program did not address how the "tight-closing" containment isolation function is performed without the assistance of the motor-operator; nor does it consider the apparent licensing basis function to close against saturated steam. Therefore, the inspectors concluded that these valves may belong within the scope of the GL 89-10 program.

Turbine Bypass Line and Steam Bypass to the Condenser Isolation Valves (MS-V2A/B and MS-V8A/B)

The UFSAR Table 10.3-1, states that the MS-V2 isolation valves have an approximate closure time of one minute against a maximum differential pressure of 1000 psid. However, the licensee's Generic Letter 89-10 program description states that their safety function is to close against approximately 51 psid and provide containment isolation following a main steamline break in the reactor building. These valves would not likely close against a differential pressure greater than approximately 500 psid.

The licensee's GL 89-10 program description indicates that valves MS-V8A/B, located downstream of MS-V2A/B, are not included in the GL 89-10 program. Valves MS-V8A/B are normally maintained in their required (open) position, but the operator does not provide an active safety function; the licensee has excluded the 8A/B valves by reliance, instead, on the upstream 2A/B valves. The inspectors concluded that these safety-related valves are the class boundary between nonsafety grade (and nonseismic) main steam line piping and may belong within the scope of the GL 89-10 program.

Emergency Feedwater Turbine Steam Supply Isolation Valves (MS-V10A/B)

The Generic Letter 89-10 program description provided a justification for removal of the EFW supply isolation valves from the GL 89-10 program, stating that: (a) automatic start of the EFW turbine is initiated by opening valves MS-V13A/B; (b) flow-through valve MS-V13 is adequate at hot shutdown, which is the ESW design basis for TMI-1; (c) valves MS-V10A/B are available during cooldown, when steam pressure is reduced to the point that MS-V13A/B no longer provides sufficient steam flow, and (d) a diverse motor-operated EFW pump will provide required flow, or MS-V10A/B could be manually opened.

c. Conclusions

The inspectors concluded that the licensee did not thoroughly document the technical bases for removing these valves from the GL 89-10 program. In addition, the inspectors considered that the licensee's justifications may have constituted changes in the design basis of TMI-1 that were not evaluated by the licensee per 10 CFR 50.59. This matter is unresolved pending additional NRC review of the design basis of the main steam valves (URI 289/96-05-02).

E4.1 TMI Response to the Arkansas One Unit 1 OTSG Blowdown

a. Inspection Scope (37551)

The inspectors reviewed the TMI response to the Arkansas Nuclear One (ANO) Unit 1 once through steam generator (OTSG) dryout event. The ANO-1 event included a stuck open main steam safety relief valve that did not re-close when steam pressure returned to normal because of a valve cotter pin and release nut that were too close

to the valve top lever. The stuck open valve resulted in the loss of the OTSG steam and water inventory. The review included the TMI response to previous OTSG dryout events, emergency operating procedures (EOPs) related to refilling a dry OTSG, plant walkdown of the TMI safety relief valves, and discussions with the OTSG system engineer and control room operators.

b. Observations and Findings

The inspectors noted that the responsible engineer was very knowledgeable about the ANO-1 safety concerns and understood how they related to the TMI OTSG safety relief valves. The engineer inspected the TMI main steam safety relief valves, prior to the inspectors review, to verify that the cotter pin was properly installed and to ensure that the release nut had at least a 1/16 inch clearance above the valve top lever. The TMI steam relief valves are Dresser valves that are identical to the ANO-1 safety relief valves. The engineer's examination confirmed that the TMI safety relief valves met the 1/16 inch vendor recommended clearance and the cotter pins were installed properly on the release nut.

The residents performed an independent walkdown to verify the physical arrangement of the cotter pins and release nuts on the safety relief valves. The inspection confirmed that the TMI safety relief valves had more than the minimum required clearance between the release nut and top lever. In fact, all valves inspected had at least one to two inches of clearance and the cotter pin was properly installed to ensure the release nut would not vibrate down if the valve lifted. The inspectors also verified that TMI had taken actions in 1984 in response to a similar problem at Davis Besse.

A second area reviewed for the ANO-1 event was the EOPs available to the operators for re-filling a dry OTSG. The inspectors discussed the ANO-1 event with the control room operators, the operators were very knowledgeable of the ANO-1 and Oconee events related to the dryout and subsequent refilling of the OTSG. The operators were very familiar with the applicable EOPs and knew exactly how to refill a dry OTSG. The procedures provided the preferred water make-up source for a dry OTSG, main feedwater (MFW) with steam pre-heating, or emergency feedwater if MFW was not available. The procedure contained excellent caution statements describing the OTSG tubesheet and shell temperature limitations and associated heatup limits. In addition, the procedure provided a clear definition of the indications a CRO would use to determine if a generator had reached a dryout condition.

c. Conclusions

The TMI engineering and operations response to the Arkansas Nuclear One OTSG dryout and stuck open safety relief valve event was comprehensive, thorough, and displayed a strong initiative to address generic safety issues at other plants in order to reduce the potential for a similar impact at TMI.

E8 Miscellaneous Engineering Issues (92903)

E8.1 (Closed) Unresolved Item 50-289/94-12-01:

The NRC Inspection Report 94-12 (URI 50-289/94-12-01) identified several concerns related to TMI's use of best-fit-straight-line calibrations with their VOTES diagnostic system. In response to this concern, TMI initiated Engineering Evaluation Request EER-94-0325, "VOTES Data Re-analysis," on October 6, 1994. This evaluation reviewed all of TMI's VOTES test data and utilized the vendor recommendations contained in Liberty Technologies' Customer Bulletin (CSB) 032, which provided the methods for applying a curve fit calibration to existing test data. Engineering Evaluation Request (EER) 94-0325 determined that operation of FW-V-92A/B should be restricted to 100 cycles because the application of curve fit calibrations increased the diagnostic inaccuracy to the extent that the measured peak thrust values exceeded 120% of the published actuator thrust ratings for these MOVs. All other reviewed MOVs were found to have acceptable torque switch settings. The Engineering Evaluation Request EER-94-0325 recommended that actuator inspections be performed and that torque switch settings be adjusted for FW-V-92A/B during the next plant outage. The inspectors reviewed September 1995 work requests that documented completion of the actuator inspection activities, including adjustment of the torque switch settings for FW-V-92A/B and confirmatory diagnostic tests. Based on review of this information, this unresolved issue was considered closed.

IV. Plant Support

R5 Staff Training and Qualification for RP&C, Security and Chemistry/Radwaste (71750)

R5.1 Personnel Self-Checking Training

a. Inspection Scope

The inspectors observed the radiological controls, security and chemistry/radwaste departments customized self-checking training. The training was initiated to reinforce department specific examples related to performance that did not meet managements' expectations and could have been prevented by the use of the "BE SURE" self-checking program.

b. Observations and Findings

The training was presented in two parts, the first segment was presented by the training department and provided an overview of the TMI self-checking program. The presentation included generic examples of cases when self-checking contributed to a positive performance and a videotape that demonstrated numerous examples of unacceptable performance due to the lack of self-checking attributes.

After the classroom presentation, the department management conducted an open forum discussion with their respective work force. Good ideas were exchanged for all of the department discussions, the most effective exchange of information

occurred in the radiological controls discussion. The Radiological Controls Director and senior managers provided their management expectations and specific department examples of less than adequate performance to the employees. The management team also fielded questions and concerns of the technicians to collect additional ideas that could result in improved plant performance. The involvement of the Radiological Controls Director and senior managers reinforced the management support and expectations to continue efforts that could lead to improved plant performance.

c. Conclusions

In conclusion, the department customized self-checking training was a positive initiative to reinforce management expectations for each department. The presentations included generic examples of cases where self-checking contributed to a positive performance and a [illegible] that demonstrated numerous examples of unacceptable performance due to the lack of self-checking attributes.

R8 Miscellaneous Radiological Control Program Items

R8.1 Previously Identified Items (92904)

(Closed) Violation (VIO, 50-289/95-13-02) Unauthorized Entry into a Radiography Area

a. Inspection Scope

On September 6, 1995, an auxiliary operator (AO) violated a radiography posting and barrier and entered an area in which radiography was being performed. The source was not exposed at the time and the AO received no exposure from the device. The violation was identified by a TMI contractor radiographer and a site engineer. The TMI radiographer and engineer did not inform either the shift supervisor or other responsible management of the radiography area boundary violation, in order to ensure adequate immediate correction of the problem, before resuming radiography operations.

b. Observations and Findings

Initially, the long-term corrective action was to review the use of postings for possible improvements. However, the licensee had not committed to any other long-term or corrective measures designed to prevent recurrence of this type of event, such as other controls and reinforcement of the requirement to adhere to radiological controls procedures and postings. In summary, the licensee's problem identification and correction process was considered inadequate, in this instance, because: (1) the shift supervisor or other responsible licensee management was not provided an opportunity to exercise management oversight and review of the occurrence prior to the resumption of radiography operations, and (2) originally determined long-term corrective actions were limited only to review of use of postings for improvements.

The licensee subsequently took several additional corrective actions to improve the radiography process. The inspectors reviewed procedure 6610-ADM-411.07, "Radiography Operations," to determine if procedure changes included a documented review of all controls for each radiography operation prior to the start of the job. The procedure was revised to provide clear direction to the radiological control personnel for proper control of radiography evolutions. The procedure was revised to include: 1) incorporation of a separate pre-job briefing checklist for radiography operations; 2) written approval from all parties in a radiography operation stating that they understand and agree to the radiological controls; 3) detailed instructions for abnormal occurrences and management review prior to resumption of radiography activities; and 4) an ALARA review by radiological engineering.

In conjunction with the procedure revision, the following additional corrective actions were performed: a review of the signs used to warn workers regarding radiography operations; revision of the training program for workers regarding these changes; and issuance of a memorandum from management that communicates management expectations regarding the use of self-checking, observation, and coaching practices.

The inspectors observed implementation of the revised radiography controls and procedure implementation during the performance of a radiography evolution in the Turbine Building on July 20, 1996. Prior to the radiography a detailed pre-evolution briefing was provided to the shift personnel by the shift supervisor (SS) and the group radiological control supervisor (GRCS). The shift supervisor and GRCS emphasized the importance of personnel accountability and reviewed some of the problems and corrective actions related to the previous radiography evolution. Plant management's decision to perform the radiography on the weekend resulted in a relatively small number of workers onsite and minimized the potential for an individual to inadvertently cross the boundaries.

In addition to the procedure changes, improved controls were used to alert plant workers to the radiography boundaries. The controls included the installation of large hexagon caution signs, 20 inches by 20 inches, and flashing strobe lights at all of the main access points. The inspectors independently verified that all access control points in the Turbine Building were posted properly prior to the start of the radiography examinations. Also, the radiological control technicians monitored the boundaries continuously throughout the evolution.

c. Conclusions

The radiography procedure changes and boundary controls significantly improved the radiological controls personnels' ability to alert all plant workers about the conduct of radiography and ensure the exclusion boundaries were controlled properly. The inspectors concluded that procedure 6610-ADM-411.07 was revised appropriately to address the prior radiography problems. In addition, the licensee's corrective actions were comprehensive and should reduce the probability of the recurrence of similar events. This item is closed.

V. Management Meetings

X1 Exit Meeting Summary

At the conclusion of the reporting period, the resident inspector staff conducted an exit meeting with TMI management on August 22, 1996, summarizing Unit 1 inspection activities and findings for this report period. TMI staff comments concerning the issues in this report were documented in the applicable report section. No proprietary information was identified as being included in the report.

X2 Meeting With GPU Nuclear Corporation Regarding Motor Operated Valve (MOV) Program Concerns

On July 22, 1996, a public meeting was held between the NRC and GPU Nuclear Corporation at the NRC Region I Office in King of Prussia, Pennsylvania. The purpose of the meeting was to discuss the TMI-1 MOV program deficiencies that were raised during an inspection held the week of June 17, 1996. GPU committed to inform the NRC about the planned corrective actions to address the MOV issues and also provide a closure date for the Generic Letter No. 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," program. The handouts used at the meeting are attached to this report.

X3 GPU Nuclear Engineering Integration Meeting

On July 24, 1996, a meeting was held between the NRC and GPU Nuclear Corporation at the NRC Region I Office in King of Prussia, Pennsylvania. The purpose of the meeting was to discuss the GPU engineering integration process and outline the newly organized engineering organization. The key elements were focused on GPU's ability to maintain an emphasis on safety as they streamline the engineering department into a more efficient unit. the handouts used at this meeting are attached to this report.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

*J. Knubel, Vice President, TMI
M. Ross, Director, Operations and Maintenance
L. Noll, Plant Operations Director
D. Etheridge, Manager, Radiological Engineering
W. Potts, Radiological Controls/Occupational Safety Director
J. Schork, Regulatory Affairs
J. Wetmore, Manager, Regulatory Affairs
G. Skillman, Technical Functions Site Director
P. Walsh, Engineering Director

* Senior licensee manager present at exit meeting on August 22, 1996.

NRC

J. Norris, TMI Project Manager, NRR

INSPECTION PROCEDURES USED

IP 62707:	Maintenance Observation
IP 62703:	Maintenance Observation
IP 71707:	Plant Operations
IP 71750:	Plant Support
IP 92903:	Followup - Engineering
IP 92904:	Followup - Plant Support

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

50-289/96-05-01, "Failure to Establish Proper Design Control Measures, when Determining the Valve Factors for certain Safety-Related Valves" (VIO)

50-289/96-05-02, "NRC review of the Design Basis of the Main Steam Valves" (URI)

Closed

50-289/95-13-02, "Unauthorized Entry into a Radiography Area"

50-289/94-12-01, "TMI's use of best-fit-straight-line calibrations with their VOTES diagnostic system"

Updated

None

LIST OF ACRONYMS USED

AB	Auxiliary Building
ALARA	As low As Reasonably Achievable
ASME	American Society of Mechanical Engineers
CDF	Core Damage Frequency
CR	Control Room
CRO	Control Room Operator
CFR	Code of Federal Regulations
DBD	Design Basis Documents
ECCS	Emergency Core Cooling System
ED	Emergency Director
EDG	Emergency Diesel Generator
EFW	Emergency Feedwater
EOF	Emergency Operations Facility
ENMCF	Event or Near Miss Capture Form
EPIP	Emergency Plan and Implementing Procedure
ESF	Engineered Safety Feature
HEPA	High Efficiency Particulate
HRA	High Radiation Area
IFI	Inspection Followup Item
IPE	Individual Plant Evaluation
IR	Inspection Report
IST	Inservice Testing Program
JO	Job Order
JPM	Job Performance Measure
LCO	Limiting Condition of Operation
LER	Licensee Event Report
MNCR	Material Nonconformance Report
NCV	Non-Cited Violation
NRC	Nuclear Regulatory Commission
NSA	Nuclear Safety Assessment
ODCM	Offsite Dose Calculation Manual
OSC	Operations Support Center
PAS	Post Accident Sample
PCR	Procedure Change Request
PPB	Part per Billion
PPM	Part per Million
PRA	Probabilistic Risk Assessment
PRG	Plant Review Group
QV	Quality Verification
RCA	Radiological Control Area
RCS	Reactor Coolant System
RP	Radiation Protection
RSP	Remote Shutdown Panel
RWP	Radiation Work Permits
SALP	Systematic Assessment of Licensee Performance
SF	Shift Foreman

SRO	Senior Reactor Operator
SS	Shift Supervisor
TI	Temporary Instruction
TLD	Thermoluminescent Dosimeter
TS	Technical Specification
TSC	Technical Support Center
UFSAR	Updated Final Safety Analysis Report
URI	Unresolved Item
VIO	Violation

TMI-1

GENERIC LETTER 89-10 MOTOR OPERATED VALVE PROGRAM

NRC / GPU NUCLEAR MANAGEMENT MEETING

JULY 22, 1996

AGENDA

I.	INTRODUCTION	R. W. Keaten
II.	DEVELOPMENT OF VALVE FACTORS	R. J. McGoey
III.	CURRENT REVIEW OF FW-V-92A/B AND RC-V-2	T. Basso
IV.	MAIN STEAM VALVES	J. Link
V.	MOTOR OPERATED VALVE INDEPENDENT REVIEW	J. C. Fornicola
VI.	SUMMARY	R. W. Keaten

INTRODUCTION

MAJOR ISSUES

- GL 89-10 Program inspection of TMI-1 June 17-21, 1996 identified areas of concern
  - Valve factor development
  - Capability of FW-V-92 A/B and RC-V-2
  - Main steam valve design basis

ACTIONS TAKEN

- Reestablished basis of original evaluations and selection of valve factors

- Engineering staff organized to reassess valve data, valve factor selection, and available valve factor

- Engineering and Safety Analysis staff rereviewed main steam system design basis.

- MOV independent review team established

RESULTS TO DATE

- New higher valve factors for valves in Groups 6 & 9 have been selected.

- Based on rereview, additional conservatisms have been incorporated for Rate of Loading and Aging Degradation.

- The valves remain capable of performing their intended safety function.

- Preliminary review of gate valves in other groups and the butterfly valves shows sufficient margin exists.

- Main steam isolation valve design bases confirmed

- Based on findings to date, a thorough technical review of the total program will be performed.

HISTORICAL DEVELOPMENT OF VALVE FACTORS

- Valve factors evolve as industry understanding and knowledge base expanded.
- Original valve factors based on accepted industry/vendor practice
- Valve factors upgraded based on early industry testing of valves similar to TMI's.
- Extensive industry search for additional valve test data initiated August 1994 to establish appropriate valve factor.
    - Reviewed EPRI valve data
    - Contacted over 90 plant/utilities (approximately 1200 man/hour effort)
    - Approximately 50 plants provided relevant data on over 500 valve tests.
    - Information collected:
        - Manufacturer
        - Disc type
        - Valve Size
        - Pressure Class
        - Material
        - Valve Factor
        - Orientation
        - Other

HISTORICAL DEVELOPMENT OF VALVE FACTORS

- Methodology Established for Valve Factor Selection

  - Hierarchy of test data significance
    - TMI valve specific test data
    - GPUN test data
    - EPRI test data
    - Applicable Industry test data
    - [PPM Model]

- Screening of Test Data
  - ~125 test results screened and included on Tables

  - Additional screening based on:
    - Data scatter concerns
    - Test methods and test accuracy
    - Differences between TMI operation and conditions of test
    - Consideration of valve/actuator orientation.

- Target Valve Factor Of 0.50 Established

  - MOV User Group and Industry Surveys
  - Used Graded Engineering Approach
  - Valves modified where margins were minimal

- Additional Valve Factor Selection Changes

RESULTS OF ASSESSING VALVE FACTOR DATA FOR GATE VALVES
DECEMBER 1994

	Valve Factor		
SOURCE OF DATA	<0.50	0.50	>0.50
GPUN		GRP 7 - 1v	GRP 8 - 1v (.7) GRP 11 - 9v (.6)
GPUN & EPRI	GRP 9 - 3v (.40) GRP 10 - 6v (.40)		
EPRI	GRP 13 - 2v (.47)		
EPRI & Industry	GRP 6 - 2v (.42) GRP 14 - 2v (.40)	GRP 1 - 3v	
Industry		GRP 4 - 4 v GRP 5 - 1v GRP 16 - 4v	GRP 3 - 2v (.6)
PPM*			GRP 15* - 2v (.555)
No data available	GRP 15 - 2v (.4)	GRP 2 - 2v	GRP 2* - 2v (.6)
No. of Valves	17	15	12

*Changes made after December 1994

1vfdata.doc

CURRENT REVIEW OF FW-V-92 A/B AND RC-V-2

OVERVIEW OF VALVE EVALUATION

- NRC questioned valve factor selection and available valve factors

- NRC pointed out misapplication of EPRI data

- We decided to take a rigorous look at valve factor selection process and rate of loading correction for available valve factor.

- With higher valve factors and more conservatism in ROL, valves remain capable of performing their intended safety function with present setup.

- Preliminary review of balance of groups completed. Similar rigorous review will be performed.

CURRENT REVIEW OF FW-V-92 A/B AND RC-V-2

VALVE FACTOR
GENERIC TMI DATA APPLICATION HIERARCHY

- TMI Valve Specific Test Data
- GPUN Test Data
- EPRI Test Data
- Applicable Industry Test Data
- PPM Model

CURRENT REVIEW OF FW-V-92 A/B AND RC-V-2

GROUP 9 (2½", 2500 # - Velan)

VALVE FACTOR DETERMINATION

RC-V-2

- Previous Valve Factor (0.400)
- TMI Valve Test data available
  - Same valve manufacturer and type
- Valve Factor (0.400) based on TMI test data
  - TMI and EPRI #13 valve test data bounded
  - ANO valve data used as reasonableness check
  - PPM not available

CURRENT REVIEW OF FW-V-92 A/B AND RC-V-2

GROUP 9 (2½", 2500 # - Velan)

VALVE FACTOR DETERMINATION

RC-V-2

- Current Valve Factor (0.472)

- Changed valve factor due to rereview of Industry and EPRI #13 valve data which included conversion of μ to valve factor and selection of a more appropriate EPRI valve factor.

- Valve Factor (0.472) chosen to bound TMI test, EPRI #13 tests and applicable Industry data.

- Calculations performed using 0.50 valve factor.

- PPM test run on TMI valve not available.

CURRENT REVIEW OF FW-V-92 A/B AND RC-V-2

GROUP 6 (6", 900 # - Crane)

VALVE FACTOR DETERMINATION

FW-V-92 A/B

- Previous Valve Factor (0.420)
  - No TMI Valve Test Data available
  - Valve Factor Data on EPRI #14 Valve
    - Valve Factor (0.420) based on EPRI #14 Valve Data (0.419)
    - EPRI Test Data is controlled and reliable
    - Consistent with applicable industry data
- PPM not available

CURRENT REVIEW OF FW-V-92 A/B AND RC-V-2

GROUP 6 (6", 900 # - Crane)

VALVE FACTOR DETERMINATION

FW-V-92 A/B

- Current Valve Factor (0.464)
    - Valve Factor changed due to conversion of EPRI #14 valve μ (friction) value to valve factor. (.419 to .436)
    - Since orientation of EPRI valve (H/V) differs from TMI valves, Industry data moved up in data hierarchy.
        - Selected valve factor (0.464) based on applicable Industry data.
            - Manufacturer, size, pressure rating, and orientation same as TMI valve.
            - Industry Valve Factor bounds EPRI #14 and applicable Industry data.
            - EPRI #14 valve provides reasonableness check.
        - PPM test data results on TMI valves yielded Valve Factor of 0.577.
- PPM tends to over-predict valve factor, for example, preliminary PPM result for EPRI #14 is 0.674 compared to test value of 0.436.

PREVIOUS VALVE CAPABILITY

Valve	PSID	Assumed Line Pressure	Selected Valve Factor	Equipment Accuracy (% Rand)	Torque Switch Repeat (% Rand)	Rate of Loading (% Bias + % Rand)	Time Related Degrad (% Bias)	Available Valve Factor
FW-V-92A	580	580	0.42	9.45	5	(2.9+10.3) ~7%	0	0.64
FW-V-92B	580	580	0.42	9.45	5	(2.9+10.3) ~7%	0	0.61
RC-V-2	2367	2367	0.40	10.08	5	(2.9+10.3) ~7%	0	0.62

CURRENT VALVE CAPABILITY

Valve	PSID	Assumed Line Pressure	Selected Valve Factor	Equipment Accuracy (% Rand)	Torque Switch Repeat (% Rand)	Rate of Loading (% Bias)	Time Related Degrad (% Bias)	Available Valve Factor
FW-V-92A	580	980	0.464	9.5	5	15	2	0.487
FW-V-92B	580	980	0.464	7.9	5	15	2	0.468
RC-V-2	2155	2155	0.472*	10.1	5	15	2	0.587

* Calculations performed with Valve Factor = 0.500

lvfdata.doc

MAIN STEAM VALVES

ISSUE

- Design basis of the isolation valves in the TMI-1 Main Steam System

TMI-1 DESIGN BASIS

- No main steam motor-operated isolation valves required to close to mitigate design basis accidents or transients.
- TMI-1 MSLB mitigated by isolation of feedwater
- Safety function for any isolation valves in main steam system
    - Provide long term containment isolation accomplished by remote manual (Control Room) operator action
    - Containment peak accident pressure (LOCA) of 50.6 psig used for MS-V-2 A/B design basis ΔP

  EOP directs operator to close MS-V-2 A/B as a prudent action in the event of an over-cooling transient - not a design basis requirement.
- No design basis requirements for MS-V-8 A/B to close.

MAIN STEAM VALVES
(Cont'd)

- Steam Line Break Licensing Basis
    - TMI-1 FSAR Chapters 10, 14, 14A
    - HELB Analyses
    - TMI-1 Restart Report -- NUREG 0680
    - NRC Operating License SER

SUMMARY

- Normally open valves-receive no ES automatic closure signal

- MS-V-2 A/B in program for long term containment isolation following MSLB inside containment.

MOTOR OPERATED VALVE (MOV)
INDEPENDENT REVIEW

MOV Independent Review Team (IRT) formed week of July 1, 1996.

Mission

- Review the GPU Nuclear MOV Program including its implementation to assess its compliance with NRC requirements and effectiveness.

Scope

This independent review includes, but is not limited to, the following:

- The specific concerns raised by the NRC, GPU Nuclear's response and associated corrective actions.
- Specific concerns identified by the GPU Nuclear self-assessment process and associate corrective actions.
- Any additional concerns identified by the team.

MEMBERSHIP

Mr. John Fornicola, GPU Nuclear Corporation - Team Leader

Mr. Paul Damerell, MPR Associates

Dr. Thomas Gerber, Structural Integrity Associates

Mr. John Hosler, EPRI

Mr. David Lewis, Shaw, Pittman, Potts, & Trowbridge

Mr. Philip Moor, GPU Nuclear Corporation

Mr. Julian Nichols, MPR Associates

Mr. Dann Smith, GPU Nuclear Corporation

Mr. Henry Stone, H. E. Stone Inc., as needed.

REVIEW TEAM PLAN

- Review applicable NRC Requirements
- Review applicable GPUN program and related documents
- Review GPUN Self-Assessment Report of the MOV Program
- Interview Personnel involved in the program
- Assess program adequacy
  - Identify potential non-compliance
  - Identify unaddressed problems/recommend corrective action
  - Recommend corrective action to prevent recurrence of similar problems

ACTIVITIES TO DATE

- Developed Charter
- Reviewing requirements/documents
- Generic Letter 89-10 and Supplements
- NSA Assessment Report 95-03
- Review NRC Inspection Reports
- TMI Program Description 89-10 MOV Program
- Requested/received several presentations from GPUN Staff
- Interviews in progress
- Requested NSA review of Engineering response to Self-Assessment Recommendations
- Requested documentation of program changes for review

PRELIMINARY OBSERVATIONS

- Self-Assessment was good in many areas, did not evaluate valve factor development in detail.
    - Some open issues
- Gaps in program documentation
    - Lack of detail in documenting certain valve factor selection and engineering judgments
- Breakdown in Communication/Teamwork between HQ and Site Engineering

REMAINING ACTIONS

- Review basis for operability of FW-V-92 A/B and RC-V-2
- Review basis for other valve factor changes
- Review responses to NSA assessment recommendations
- Complete interviews
- Review OC MOV Program
- Issue final independent review report

SUMMARY

Valve Status

- Upgraded valve factors for FW-V-92 A/B and RC-V-2 and additional conservatisms for Rate of Loading and Aging Degradation have been included in the program.
- Magnitude of Group 6 & 9 valve factor changes indicates other valve groups have sufficient margin to accommodate similar changes.
- Main steam isolation valve design basis confirmed.
- Review to date indicates that program valves are capable of performing their safety function.

Additional Actions

- Detailed design review of entire program to be performed to ensure adequacy in margin for all program valves.
- Continue to evaluate methods to increase valve margins where appropriate.
- Independent review of the historical and current MOV program is intended to provide adequate assurance that problems are resolved and to recommend corrective actions to prevent recurrence.

AGENDA

July 24, 1996

A. Introduction
T. G. Broughton

B. Engineering Integration Process
R. W. Keaten

C. Materials and Services Organization Restructuring
J. Langenbach

D. Summary
T. G. Broughton

Engineering Integration Process

Initiative for Engineering Integration Process
Process Focus

Process focus started in 1994 to identify and define core business processes with the goal of improving the efficiency and effectiveness of processes.

Review looked broadly across GPU Nuclear processes supporting plant operation and looked more closely at the engineering and technical services processes.

Key Elements:

- Identify opportunities to streamline and standardize workflow without compromising quality.
- Evaluate process and end products against identified desired characteristics and benchmark against other companies.
- Identify and eliminate duplicate and overlapping capabilities.
- Evaluate various engineering activities.
    - Technical Functions
    - Plant Engineering
    - Plant Operations Engineers
    - Plant Maintenance Assessment
    - Radiological Engineering

Engineering Integration Process
Engineering Integration Team

- Need for integration was emphasized in revised organizational principles forged by Senior Staff in February 1996.

- Engineering Integration Team (EIT) constituted March 1996.

  - Identify and eliminate unneeded engineering-related work.
  - Simplify and streamline necessary work.

Objectives:

- Focus on revised organization principle which commits to a strong integrated engineering function

- Continue to provide day-to-day engineering support for Operations and Maintenance.

- Continue to provide Long Range Plant System, and Component Engineering Planning and Development.

- Reduce nuclear plant operating costs

Mission Statement

Considering the challenges of deregulation, broadly examine the engineering and technical activities performed in support of Oyster Creek and TMI. Recommend changes to process, work activities and structure which will reduce cost and align resources in a seamless manner, while maintaining high levels of quality and safety.

Engineering Integration Process Approach

- Focus Groups were conducted to identify opportunities and critical success factors to be used in restructuring engineering.

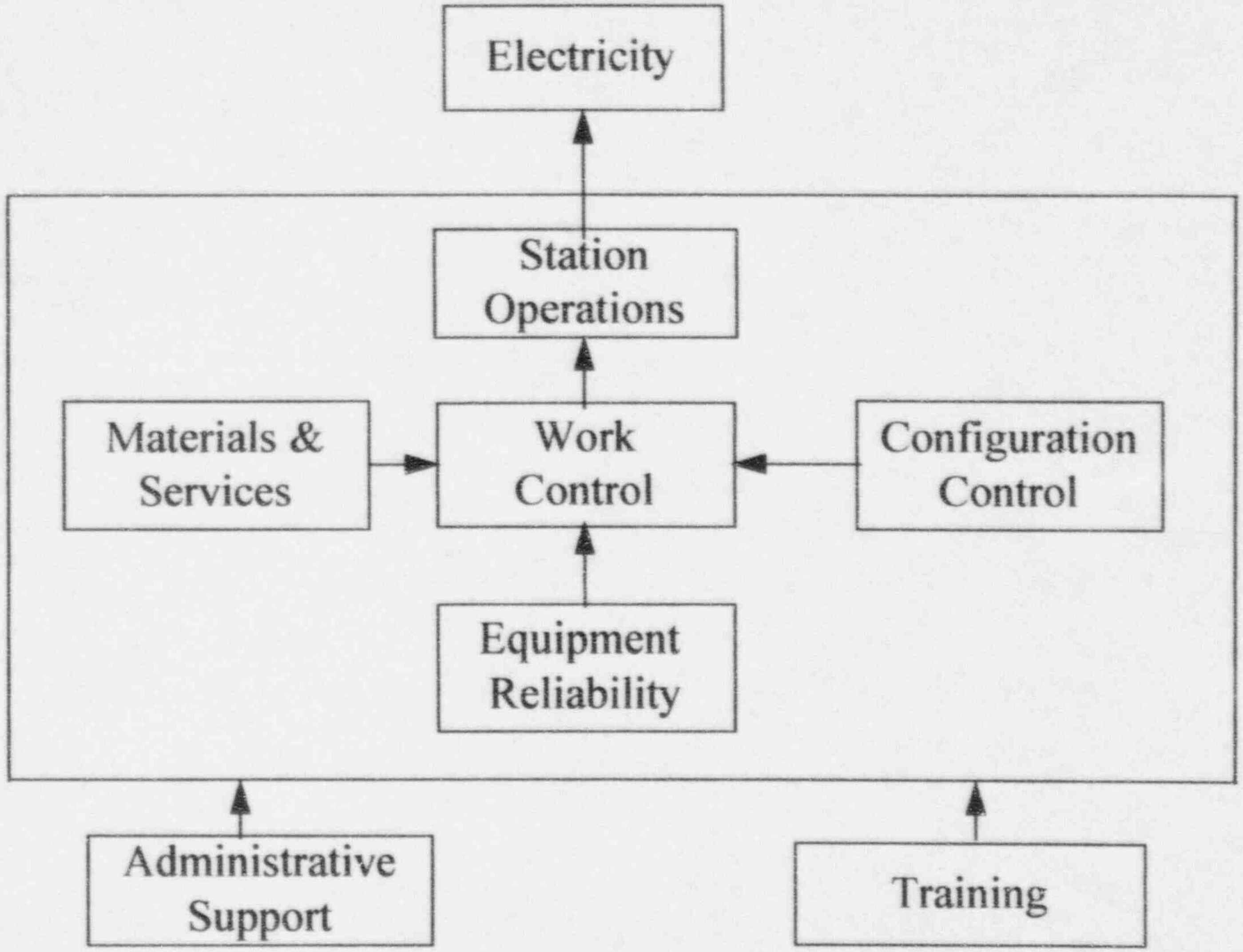
- Key stake-holders were interviewed to confirm needs and expectations.

- Electronic Mail system was set up to receive comments and suggestions.

- Two EITs were established to analyze engineering work and map the work flow.
  - Functional Methodology
  - Process Methodology

- After internal reviews were completed, the teams visited low cost/high performance utilities (SALP Ratings) to benchmark with their engineering organizations.

INPO facilitated development of Processes under industry's
Stragetic Plan for Building New Nuclear Power Plants.



process.ppt

Engineering Integration Team Results

After all internal and external reviews were completed, the two sub-teams compared results - the process evaluation was found to give the most useful insights.

The team unanimously recommended that:

- All engineering functions be integrated into a new Engineering Division.
- The new division be organized along the lines of the industry's Equipment Reliability and Configuration Control processes.
- Functional groupings be used as appropriate.
- The engineering "center of gravity" be shifted to the plant sites.
- Some engineering functions remain centralized.

Equipment Reliability (ER) Process
Key Features and New Ideas

- ER Process consolidated under single organization at each site which includes System Engineers, Component Engineers, and ER Program Owners.

- System Engineers become System "Owners" - provide single point accountability for all system issues.

- System Engineers:

  - Maintain existing System Engineer accountabilities.
  - Resolve long-term and short-term issues affecting the reliable performance of the system.
  - Assume ownership of P. M. Program technical content for their systems.
  - Assume ownership of on-line maintenance work technical content.
  - Perform SU&T functions on their systems.
  - Coordinate and chair system performance team.

- System Engineers draw on Component Engineers, site based Design Engineers, and Corporate based Specialists to resolve issues. Centers of Excellence maintained.

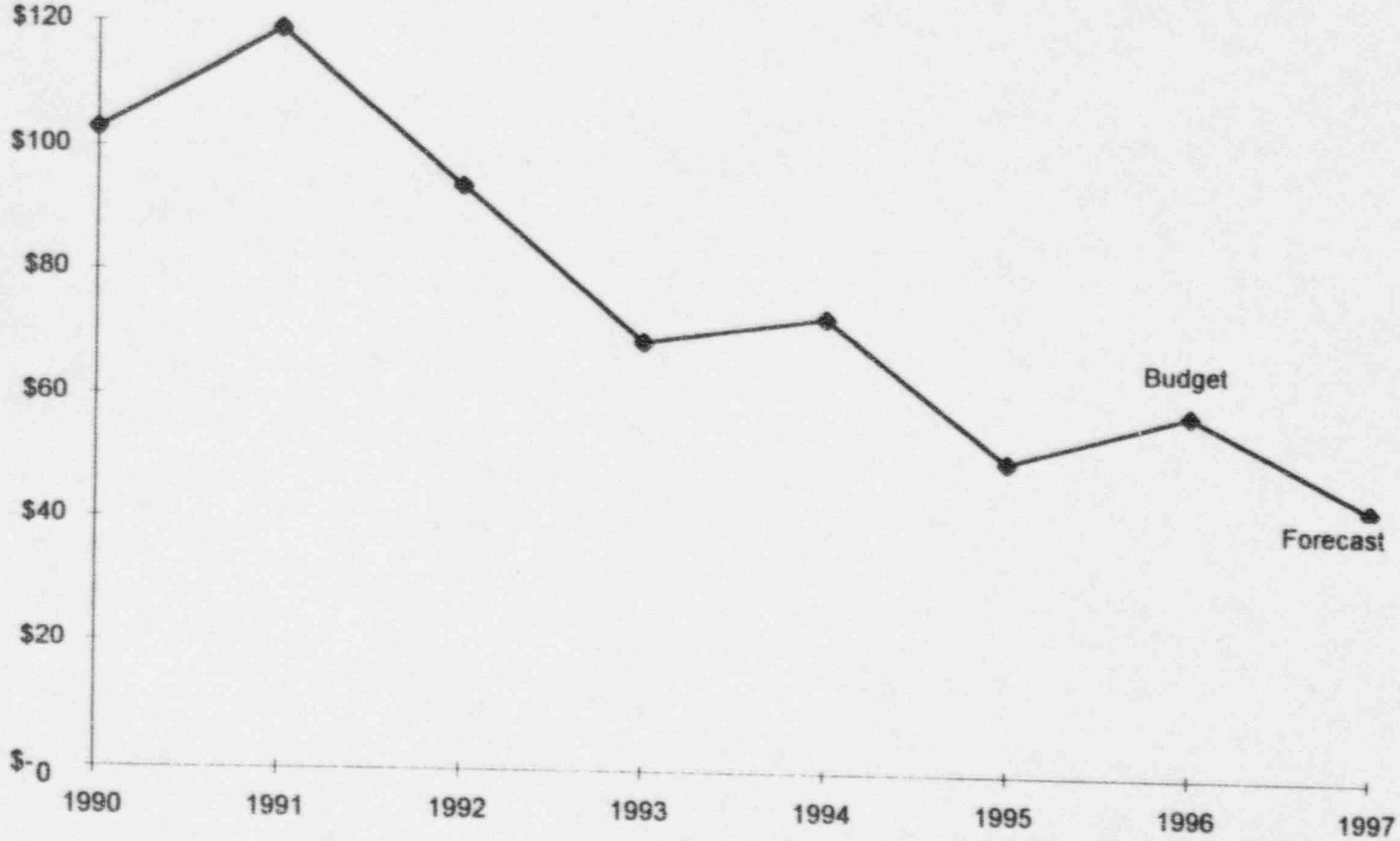
Configuration Control (CC) Process
Key Features and New Ideas

- Most modifications performed by site based Design Organization

- Design & Project Management Centers of Excellence maintained on site.
  - Fewer plant modifications and reduced engineering workload have resulted in reduced expenditures.

- Configuration Maintenance group supports procurement and document control activities.

GPU NUCLEAR CAPITAL SPENDING
($ IN MILLIONS)



Center of Excellence

Objective:

Retain control of proficiency and assurance of work standards.

- Accountabilities will be assigned to appropriate departments/personnel to ensure processes and procedures meet all required technical and regulatory standards.

- Personnel who perform work with these processes and procedures are trained, qualified, and perform the technical and regulatory work standards.

EXAMPLE: Startup and Test sub-process

- Current Start-up and Test function will be included in the Equipment Reliability Process, and System Engineering sub-process.

- System Engineering Managers will own Start-up and test process and assure work is performed.

- Qualified and certified Start-up and Test Engineers will perform the work.

<u>Selection Process for Engineering Division</u>

- Key importance in selection process is GPU Nuclear's need to maintain excellence in its primary focus on nuclear safety, power production, and business acumen.

- Senior Level Selection Committee established to select new Directors.

- Targeted Selection Process focused on "dimensions" - The key characteristics required for leaders in the new organization.

- Selection based on:

  - Results of interviews using behavior-based questions for each dimension.
  - Education and Experience.
  - Prior Performance.

- New Directors met to finalize organization structure and begin selection process for managers.

Selection Process for Engineering Division Continued

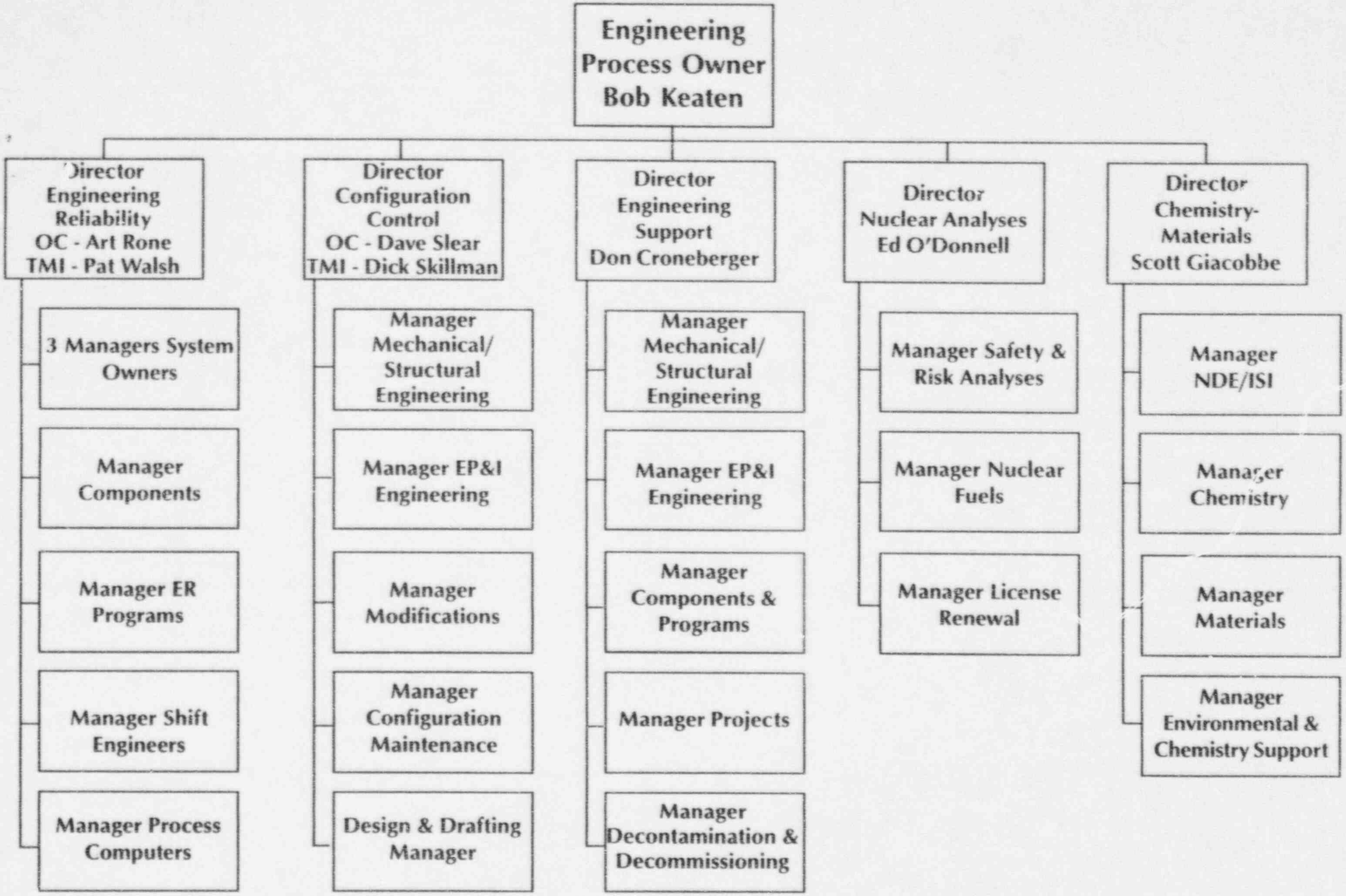
- Individual managers were selected utilizing a Selection Committee and the Targeted Selection Process.

- Selected managers met to finalize initial staffing levels and plan remaining selection activities.

- Individual contributors are being reselected by selection committees based on appropriate criteria including key behavioral dimensions.

- Engineering Division staffing is based on retaining the best qualified personnel.

Engineering Process Management Structure



Engineering Division Staffing Reductions

- Approximate 20% staff reduction overall

  - Engineering totals today approx. 480
  - New Engineering Division approx. 390

Transition Plan

Objective:

Effect a complete, orderly, and systematic transition from the current Engineering Organizations to the new Engineering Division.

- Transition team formed to implement new GPU Nuclear Engineering Organization.

- Team is comprised of internal experts who have the lead for key activities:

    - Procedures and processes
    - Budget and cost
    - Training
    - Information Technology and Data Systems
    - Process Mapping and Center of Excellence
    - Facilities
    - Nuclear Safety Assessment, Self-Assessment, and Process Measures
    - Organizational Document Revision
    - Regulatory Compliance Documents

Transition Plan
Continued

- A certification process will be used to track, document, validate, and certify all work has been accomplished.
  - Process is similar to that used during the TMI Unit 2 merge with TMI-1; The Site Services Division merge into the Oyster Creek Division; and Restart Certification from plant refueling outages.
- A database has been developed to inventory all work tasks being performed by current Engineering personnel.
- Work will be reassigned to appropriate organizations and individuals.
- Discontinued work will be justified and documented.
- NSA will verify all Operational Quality Assurance Plan requirements are met.

Self-Assessment Engineering Organization

- Importance of effective self-assessment program recognized and included in Corporate 1996 Safety Goals.

- Company-wide standard under development.

- Includes departmental assessment and independent oversight aspects.

- Engineering Transition Team will develop specific program for Engineering

- Product to be a model for other departments.

Implementation of Integrated Engineering Organization

- Completion of the new process organization is targeted for early August 1996.

- Goal is full implementation of the new organization before the OCNGS outage begins in September 1996.

Materials and Services Organization

Restructuring

Materials Management Reorganization to Materials and Services

- Reasons for Reorganization

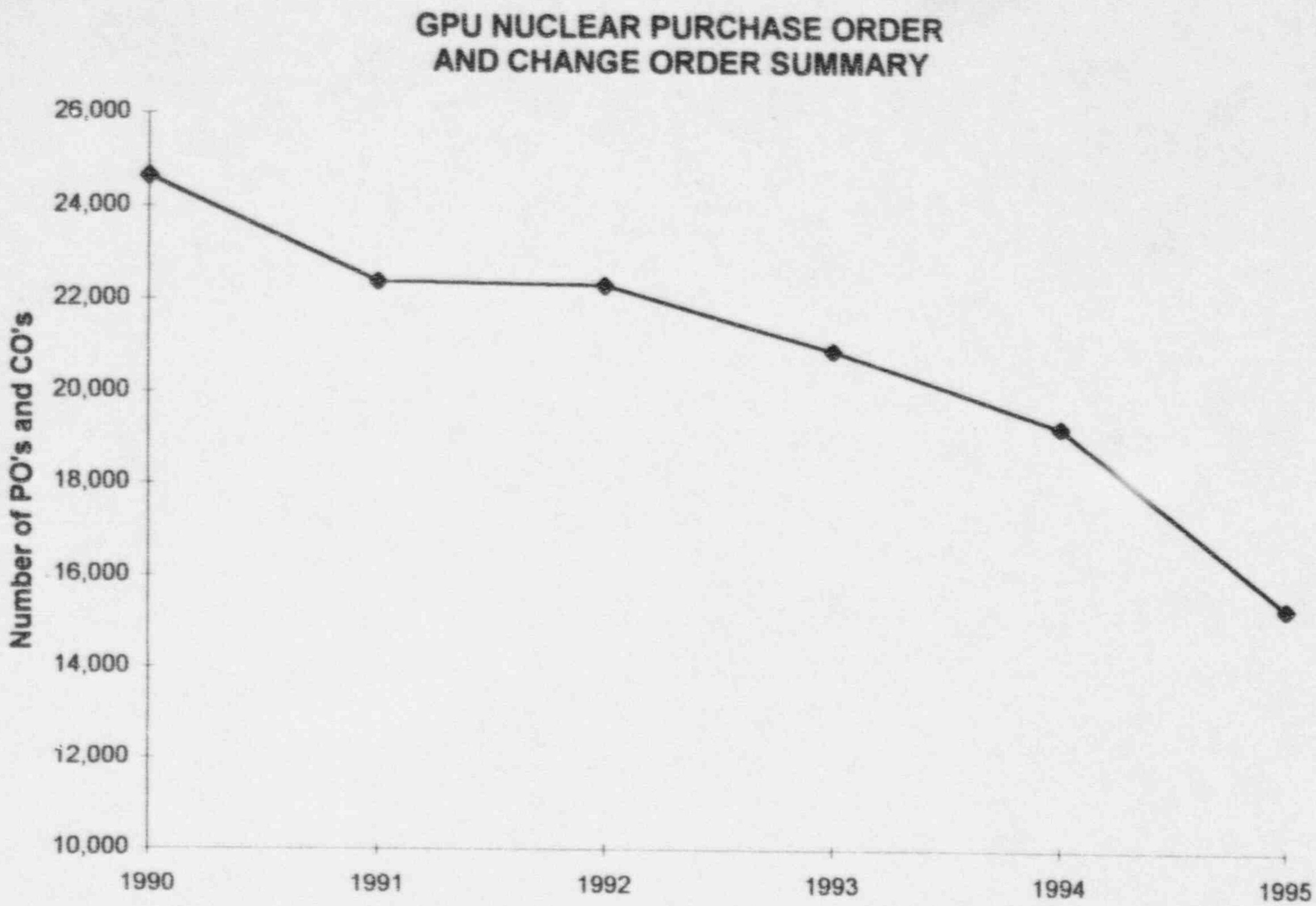
  - Declining Workload
    - Fewer Capital Modifications
    - Longer Term Agreements (Partnering)
    - Leveled Workload (More work done while operating)

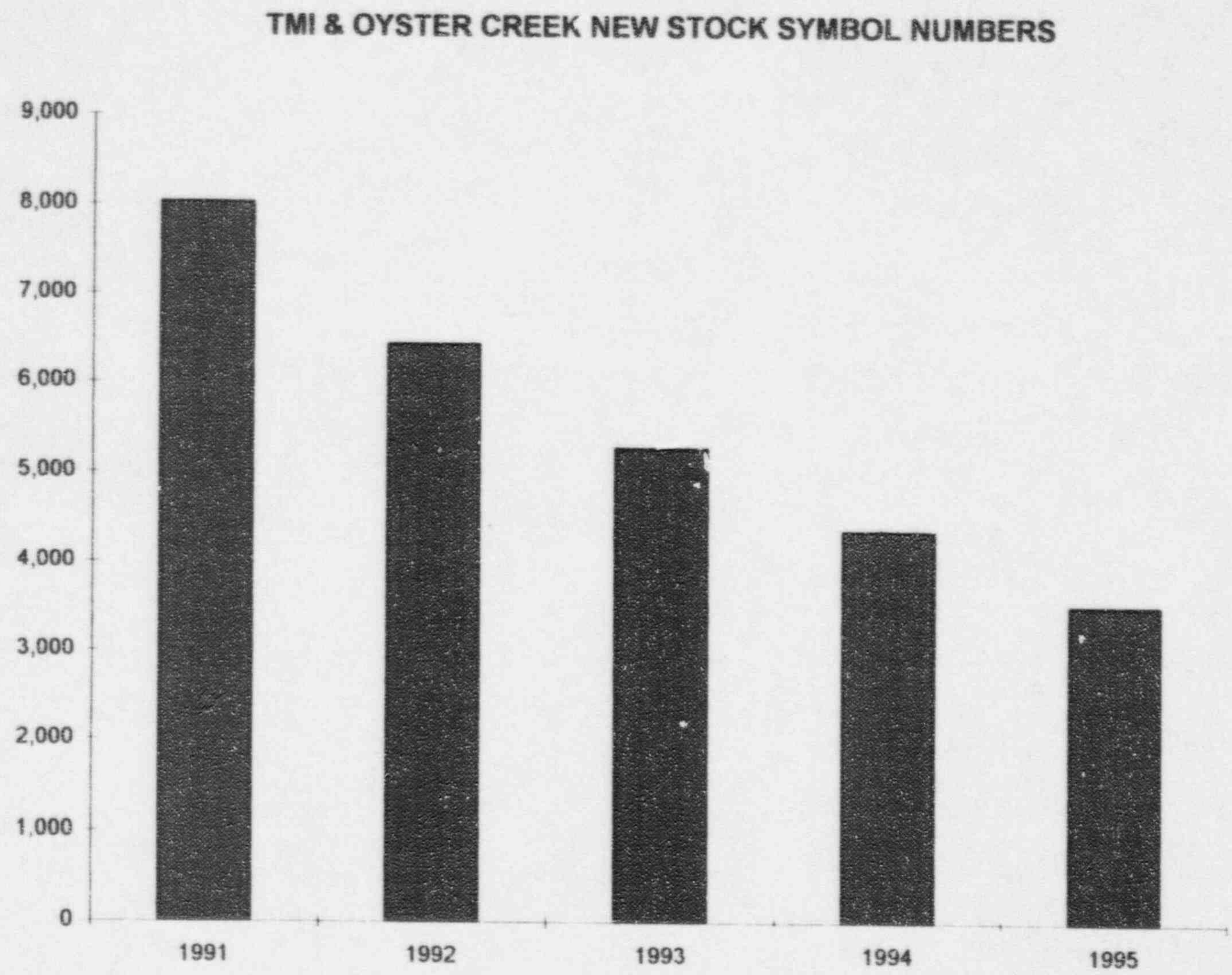
  - Materials Management Process Reengineering Recommendations
    - Process Focus
    - Broader Span of Control

  - Low Risk
    - Minimal Nuclear Safety Implications
    - Minimal Operational Impact Potential
    - Nuclear Safety Assessment oversight

- Future efficiencies expected to be gained from process focus

- Initial changes - Approximately 20 fewer positions out of 100 full time employees
  - Went from 16 to 8 managers and supervisors
  - Average span of control from 6 to 10


GPU NUCLEAR PURCHASE ORDER
AND CHANGE ORDER SUMMARY
Number of PO's and CO's
26,000
24,000
22,000
20,000
18,000
16,000
14,000
12,000
10,000
1990
1991
1992
1993
1994
1995


TMI & OYSTER CREEK NEW STOCK SYMBOL NUMBERS
9,000
8,000
7,000
6,000
5,000
4,000
3,000
2,000
1,000
0
1991
1992
1993
1994
1995

Summary

- New Engineering Support and Materials & Services organizations are based on Process Model Methodologies.

- These models provide more cost-effective support organizations while maintaining primary focus on nuclear safety and safe plant operation.

- Elimination of low value, redundant work activities and organizational overlap as well as increased spans of control, single points of accountability, and staffing at levelized requirements achieve substantial efficiencies.

- Staff reductions supported by declining work loads and reduced plant modifications.

- Nuclear Safety Oversight Groups will monitor the implementation of the new organization.

- Strengthened and formalized the self-assessment program to monitor work quality and other performance measures.

- Transition program established.

- Process review for other areas will begin later this year.

    - Maintenance / Work control
    - Operations

- Encourage NRC feedback