GPU NUCLEAR CORPORATION

OYSTER CREEK NUCLEAR GENERATING STATION

Operating License No. DPR-16
Docket No. 50-219
Technical Specification Change Request No. 216

This Technical Specification Change Request is submitted in support of the Licensee's request to change the Appendix A Technical Specifications to Operating License No. DPR-16 for Oyster Creek Nuclear Generating Station. As a part of this request, the proposed replacement pages for Appendix A are also submitted.

GPU Nuclear Corporation

John J Barton

John J. Barton
Vice President and Director
Oyster Creek

Sworn and Subscribed to before me this 15th day of June, 1994.

Judith M. Crowe

A Notary Public of New Jersey

JUDITH M. CROWE
Notary Public of New Jersey
My Commission Expires 1/25/95

Oyster Creek Nuclear Generating Station (OCNGS)
Operating License No. DPR-16
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1.0. PROPOSED TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR)

GPUN requests that the following pages of the OCNGS Technical Specifications (Tech. Specs.) be replaced as indicated below:

Replace Page: 2.3-2.

2.0. DESCRIPTION OF CHANGES

Tech. Spec. 2.3.D, "Reactor High Pressure, Relief Valve Initiation" is revised to increase the Limiting Safety System Setting by 15 psig. This will result in two (2) Electromatic Relief Valves (EMRVs) initiating ≤ 1085 psig and three (3) EMRVs initiating ≤ 1105 psig.

3.0. DISCUSSION OF THE REASONS FOR CHANGE, AND SAFETY IMPLICATIONS

The purpose of this TSCR is to propose the increase in the Technical Specification specified high pressure actuation setpoints of the Electromatic Relief Valves (EMRVs) by 15 psig. This will result in the increase of the Tech. Spec. 2.3.D specified EMRV high pressure actuation setpoints for: (a) two (2) EMRVs at ≤ 1070 psig to ≤ 1085 psig; and, (b) three (3) EMRVs at ≤ 1090 psig to ≤ 1105 psig.

The "Bourdon tube" type pressure switches currently in use at Oyster Creek experience drift, which results in exceeding the existing "as-found" setpoints. The direction and amount of this drift has been determined by ongoing re-evaluations of switch performance. Increasing the Tech. Spec. 2.3.D specified setpoints by 15 psig will provide for expanding the 'as-found' tolerance bands. Increasing these tolerance bands should serve to ensure that the setpoints will remain within the Technical Specification requirements over a nominal 24 month operating cycle. The "as-left" setpoints and tolerances of the current plant operating procedures are not affected by this change request.

Analyses and evaluations (Reference 6.1) have been performed to demonstrate the acceptability of the 15 psig increase.

Those analyses and evaluations include:

(1) the determination of the continued operability of the plant response to over-pressurization transients at the higher setpoints using the NRC-approved GPUN reload analysis methodology;

(2) the determination of the continued acceptability of structural response of the EMRV Main Steam Line branch connection, EMRV discharge header, "Y-quencher", torus shell, torus attached piping and torus internal structures to EMRV actuation as previously determined during the conduct of the Mark I Containment Long-Term Program;

(3) a detailed review of the short term actions taken as part of mitigating the structural response of the torus shell, to EMRV actuation, to determine which of those short term actions, if any, continued into the bases of the Mark I Containment Long-Term Program; and,

(4) a detailed review of the past EMRV related industry issues and the current Oyster Creek NRC safety evaluation reports (SERs) associated with major reevaluations and/or upgrades such as those associated with NUREG-0737; 10 CFR 50.49; 10 CFR 50, Appendix R etc. to ensure that all EMRV high pressure actuation setpoint dependent safety considerations were addressed.

The analyses and evaluations, however, did not include Anticipated Transient Without Scram (ATWS) because previous evaluations have demonstrated that the ATWS events are bounded by the Main Steamline Isolation Valve closure transient - for safety valve sizing performance evaluation. The analysis that supported License Amendment No. 150 demonstrated that an MSIV closure ATWS (8 safety valves) with recirculation pump trip (RPT) and EMRV actuation was bounded by the MSIV closure with High Flux Scram (9 safety valves) and no RPT or EMRV actuation. That analysis addressed the over-pressurization limits associated with an ATWS event, and demonstrated that the ATWS transients do not have to be re-analyzed for each reload. Since the cycle 14 reload transients were re-analyzed at the proposed TSCR 2.3.D high pressure EMRV actuation setpoints, using the GPUN reload methodology, the ATWS transients were not re-analyzed.

The results of these analyses and evaluations are as delineated below:

Overpressurization Events

Pressurization transients which result in EMRV actuation were re-analyzed using the higher EMRV high pressure actuation Tech. Spec. setpoints. These analyses are the same as those previously re-analyzed as part of TSCR No. 98 which resulted in License Amendment No. 62. Amendment No. 62 authorized the elimination of the 25 psig margin between the peak pressure realized during a Turbine Trip Without Bypass transient and the setpoints of the Reactor High Pressure Safety Valves.

Turbine Trip With Bypass Transient:

A turbine trip is the primary turbine protection mechanism and is initiated whenever various turbine or reactor plant malfunctions threaten turbine operations. The turbine trip initiates closure of the turbine stop valves and opens the bypass valves. Upon initiation of the trip, pressure rises, opening the first two EMRVs at 1.86 seconds and the next three EMRVs at 2.31 seconds. A peak steam dome pressure of 1147.6 psia occurs at 2.44 seconds. The delta-CPR for this transient is bounded by the delta-CPR associated with the Turbine Trip Without Bypass transient.

Turbine Trip With Bypass (Current EMRV setpoint +15 psi) Transient:

The turbine trip initiates closure of the turbine stop valves and opens the bypass valves. Upon initiation of the trip, pressure rises, opening the first two EMRVs at 2.24 seconds and the next three EMRVs at 2.53 seconds. A peak steam dome pressure of 1155.5 psia occurs at 2.62 seconds. The delta-CPR for this transient is bounded by the delta-CPR associated with the Turbine Trip Without Bypass transient.

Turbine Trip Without Bypass Transient:

The turbine trip without bypass transient (TTWOBP) is the overpressure transient normally analyzed for each Oyster Creek reload. It produces the most limiting Critical Power Ratio (CPR) for overpressurization events. The failure of the bypass valves to open following a turbine trip results in a rapid pressurization, opening the first two EMRVs at 1.728 seconds and all five valves by 1.994 seconds. A peak pressure of 1284 psia (recirculation line pressure) is reached in 4.3 seconds. Pressure in the steam dome reaches 12[illegible] psia. Initial CPR was 1.534 and 1.481 for the GE 7 and GE 8 fuels respectively. The delta-CPR for this transient is 0.314 and a Minimum Critical Power Ratio (MCPR) occurs at 1.06 seconds, i.e., prior to the EMRVs opening.

Turbine Trip Without Bypass Transient (Current EMRV setpoint +15psi):

The failure of the bypass valves to open following a turbine trip results in a rapid pressurization, opening the first two relief valves at 1.873 seconds and all five valves by 2.158 seconds. A peak pressure of 1290 psia (recirculation line pressure) is reached at 5.2 seconds. Pressure in the steam dome reaches 1269.2 psia. Initial CPR was 1.534 and 1.481 for the GE 7 and GE 8 fuels respectively. The delta-CPR is the same as with the lower EMRV setpoints since MPCR occurs before the EMRVs open.

Safety Valve Sizing Transient:

The main steam line isolation closure transient demonstrates the adequacy of the spring loaded code safety valves in providing overpressure protection. The results of this transient analysis are not sensitive to the EMRV overpressure actuation setpoints. This is because the EMRVs, as well as the isolation condenser, are considered inoperable for purposes of this analysis.

Main Steam Line Isolation Closure Transient:

This transient results from a 3.0 second closure of the main steam isolation valves. A reactor trip occurs at 10% closure of the MSIV at 0.5 seconds. The first two EMRVs open at 3.59 seconds and the next three EMRVs open at 3.86 seconds. A peak dome pressure of 1194.5 psia occurs at 4.8 seconds. The delta-CPR for this transient is bounded by the delta-CPR associated with the Turbine Trip Without Bypass transient.

Main Steam Line Isolation Closure (Current EMRV setpoint +15psi) Transient:

This transient results from a 3.0 second closure of the main steam isolation valves. A reactor trip occurs at 10% closure of the MSIV at 0.5 seconds. The first two EMRVs open at 3.79 seconds and the next three EMRVs open at 4.05 seconds. A peak dome pressure of 1202.2 psia occurs at 4.7 seconds. The delta-CPR for this transient is bounded by the delta-CPR associated with the Turbine Trip Without Bypass transient.

Feedwater Controller Failure Transient:

This transient represents the plant response to the failure of the feedwater control system in the maximum demand position with initial reactor vessel level at the low alarm setpoint. The maximum feedwater flow used is 120% of rated flow. The transient proceeds by ramping the feedwater flow from its rated value to 120% in 0.1 seconds with the feedwater controller disabled. The increase in core inlet subcooling due to the increase in feedwater flow causes positive reactivity insertion which leads to a power increase. The downcomer water level will increase due to the steam/feedwater flow mismatch until the high water level turbine trip setpoint is reached when a turbine trip is initiated through a turbine stop valve (TSV) closure. The reactor vessel high water level setpoint is reached after 32.6 seconds.

The bypass valves open after the TSV closure and the first two EMRVs open at 34.65 seconds followed by the remainder three EMRVs opening at 34.96 seconds. A peak pressure of 1177.7 psia (recirculation line pressure) is reached at 35.1 seconds. Pressure in the steam dome reaches 1148.5 psia. The delta-CPR for this transient is 0.232 and a MPCR occurs at 33.6 seconds, i.e., prior to EMRV opening.

Feedwater Controller Failure (Current EMRV setpoint +15psi) Transient:

The transient is initiated by ramping the feedwater flow from its rated value to 120% in 0.1 seconds with the feedwater controller disabled. The increase in core inlet subcooling due to the increase in feedwater flow causes positive reactivity insertion which leads to a power increase. The downcomer water level will increase due to steam feed mismatch until the high water level turbine trip setpoint is reached when a turbine trip is initiated through a TSV closure. The high water level setpoint is reached after 32.6 seconds. The bypass valves open after the TSV closure and the first two relief valve open at 34.89 seconds followed by the remainder three EMRV at 35.18 seconds. A peak pressure of 1182 psia (recirculation line pressure) is reached at 35.2 seconds. Pressure in the steam dome reaches 1153.8 psia. The delta-CPR is the same as with the lower EMRV setpoints since MPCR occurs before the EMRVs open.

Loss of Electric Load Transient:

The loss of electrical load transient exhibits the characteristics as the turbine trip with bypass transient. However, since the steam flow to the turbine is initially reduced by action of the throttle valve, the pressure and power transients are milder. A re-analysis of this transient was not performed, since it is bounded by the turbine trip with bypass transient results.

Loss of Main Condenser Vacuum Transient:

The sudden loss of main condenser vacuum causes a simultaneous scram, turbine trip, and bypass valve closure. This transient is bounded by the turbine trip without bypass transient.

Loss of Auxiliary Power Transient:

Loss of auxiliary power causes the loss of condenser cooling water, a trip of feedwater pumps, a trip of the recirculation pumps and a turbine trip. It is assumed that the turbine bypass valves will remain open for about 1.5 seconds. The turbine bypass valves trip when the main condenser vacuum reaches 10 inches Hg. Reactor operating experience has shown that vacuum does not drop below the 10 inch setpoint until after 1.5 seconds. This period of steam bypass minimizes the magnitude of the reactor power transient following the turbine valve closure, and the ensuing pressure transient is less severe than that obtained in the turbine trip without bypass transient (TTWOB). Since the results are bounded by the TTWOB, this transient is not re-analyzed.

Mark 1 Containment Long-Term Program

The changes to the stresses and fatigue usage previously determined as part of the original "Oyster Creek Nuclear Generating Station Mark I Containment Long-Term Program Plant Unique Analysis," (OCPUA), (References 6.2 through 6.6) were re-evaluated by MPR Associates Inc. utilizing a three step process (Reference 6.7). The definition of the loads associated with EMRV actuation was originally performed with all five (5) EMRVs at a set pressure of 1070 psig. To support this Technical Specification Change Request, a set pressure of 1105 psig for all five (5) EMRVs was used. The results of the OCPUA had determined that simultaneous actuation bounded sequential actuation and it was decided to maintain consistency with these original results.

Raising the EMRV setpoint to 1105 psig increases the EMRV actuation induced loads by less than 3.3%. This 3.3% factor was derived by analysis (Reference 6.7). An additional 0.5% had to be added to this factor to account for a minor error found in the OCPUA Reports resulting in a correction to the design steam flow rate used in the original reports. The total 4.0% increase was then used to support the proposed EMRV actuation setpoints increase. The EMRV actuation induced loads include:

(1) transient pressure and thrust loads on the EMRV discharge header;

(2) transient pressure and thrust loads on the EMRV discharge header discharge device (i.e. the Y-quencher);

(3) water jet loads on torus submerged structures;

(4) air bubble pressure oscillation loads on the torus shell; and,

(5) air bubble drag loads on torus submerged structures.

The review process utilized to determine the acceptability of raising the setpoints to 1105 psig consisted of:

Components which were originally reported to be below allowable stress and/or load in the OCPUA were increased by 4.0% irrespective of EMRV contribution to the load. If these components were still below allowable with the 4.0% increase in load, they were considered acceptable. The majority of components fall into this category. After applying the 1.04 factor to all stresses and loads reported in the OCPUA, a total of ten outliers were identified in which the factored stress of load exceeded the OCPUA allowable. For eight of these ten outliers, stress or loads were found to be acceptable by examining the OCPUA calculations and determining the impact of EMRV load increase by increasing only the portion of the total stress or load that was due to EMRV, as opposed to increasing the original OCPUA load values.

Mark 1 Containment Long-Term Program - Cont'd.:

At this point, the EMRV discharge piping itself, and the vent line/vent header intersections remained to be dispositioned. For these two components, the stresses reported in the original OCPUA analyses (References 6.2 - 6.5) slightly exceeded allowable values for several load combinations, but were considered acceptable due to conservatism in the original analysis methods.

To determine the acceptability of increasing the EMRV activation-induced loads on these components, the stress analyses performed were reviewed to determine whether more realistic (less conservative) analysis methods could be used to more accurately determine the stresses for these components.

In these cases, using the square root sum-of-the-squares summation method (SRSS), as approved by the NRC in NUREG-0484 (Reference 6.8), for independent dynamic loads (such as earthquake, EMRV discharge and loss of coolant accidents), results in stresses that are within allowables for the increased EMRV discharge loads as well as for the original Mark I Containment Long-Term Program loads.

SRSS summation of independent dynamic loads was used previously in MPR Report, MPR-772, (Reference 6.5). Use of SRSS resulted in stresses below allowable for the increased EMRV discharge loads for the vent line/vent header intersection load combinations affected by EMRV. Also, use of SRSS resulted in stresses considered acceptable for all but one location on the EMRV discharge piping (i.e., stresses were equal to or less than stresses considered acceptable in the original OCPUA analyses). This one location, the connection between the main steam line and the south header discharge line, was found to be stressed to 21.66 ksi for the load case consisting of EMRV discharge plus deadweight loads. This stress is about 3.1% above the OCPUA allowable stress of 21.0 ksi. (Use of SRSS for this combination did not reduce the stress since there was only one dynamic load.)

Because of the conservatisms built into the OCPUA load definition for EMRV discharge, and recognizing the conservatisms built into the code stress limits, this currently calculated overstress by 3.1% is considered acceptable. For example, the EMRV flow calculation as specified by the NUREG-0661 includes a 5% general purpose conservatism. In the OCPUA report for the torus attachment piping, calculated stresses for other load combinations in portions of the EMRV line exceeded allowables by as much as 7.3%. Because of the conservative approaches taken in the OCPUA, this overstress was judged to be acceptable, as indicated in MPR-734, which was submitted by GPUN to and accepted by the NRC in 1983 and 1984 respectively.

Mark 1 Containment Long-Term Program - Cont'd.:

The impact of the EMRV load increase, resulting from the setpoint increase, on calculated fatigue usage was determined to be an increase in that usage by a factor of about 1.11. Consequently, the fatigue usages reported in the OCPUA were checked. It was determined that the total usage would not exceed the allowable total usage of 1.0. This evaluation of the increase in stresses and fatigue usage resulting from the increase in EMRV setpoints confirmed that the structural response of the EMRV branch connection to the Main Steam Line, EMRV discharge header and "Y-quenchers", torus shell, torus attached piping and torus internal structures to EMRV actuation as determined consistent with the Mark I Containment Long-Term Program methodologies remains acceptable.

EMRV Actuation - Short Term Considerations:

Reportable Occurrence No. 50-219/76-29-1P (1/3/77), as supplemented by JCP&L letter to the NRC, EATJM-29 (1/10/77), reported the results and actions taken in response to the results of testing conducted in November, 1976. That testing was conducted to measure the structural response of the torus to EMRV actuation (i.e.. "steam vent clearing" as it was termed at the time). Those results indicated that the structural loading on the torus was unacceptably high during simultaneous actuation of EMRVs in the same header and that higher than desired water slugs could be expelled during subsequent EMRV actuation. During this testing the EMRV discharge device was a "canal fitting."

To provide short term mitigation of these effects the following actions were taken and modifications were performed:

1. staggered overpressure actuation set and reset setpoints were established;

2. addition of redundant two minute timers in each independent ADS channel;

3. addition of an ADS actuation staggering relay in each ADS channel for EMRVs NR-103B, NR-103C and NR-103E;

4. addition of "valve open plus ADS" logic for EMRVs NR-103A and NR-103D;

5. addition of increased EMRV discharge header vacuum breaking capacity;

EMRV Actuation - Short Term Considerations (Cont'd.):

6. replacement of the "canal fitting," as the EMRV discharge header discharge device, with the "Y-quencher";

7. reduction of the isolation condenser initiation time delay to 0 seconds (currently $\leq$ 3 seconds); and,

8. reduction of the turbine trip anticipatory scram from 40% of rated power to 0% (currently 10%).

Testing of the structural response of the torus to EMRV actuation with the "Y-quencher" was conducted in August, 1977 (Reference 6.2).

The comparison of these results to those of the November, 1976 tests (JCP&L Letter No. EATJM-29, dated 1/10/77) established that:

1. torus shell stresses resulting from EMRV actuation were reduced by a factor of 20 to 30;

2. vent header support column loads were reduced by a factor of 7;

3. torus shell deflections near the Y-quencher were reduced by a factor of 16 from those measured near the replaced canal fitting;

4. the deflections at the core spray suction header nozzle were reduced to zero; and

5. the increased capacity of the vacuum breakers prevented the long water slug condition which occurred in the November 1976 test.

This testing (Reference 6.2) constituted the input into the original OCPUA associated with the Long Term Program for definition of loads associated with EMRV actuation. The installation of the Y-quencher, increased vacuum breaking capacity, installation of the torus saddles and hoop straps and an assumption associated with the time interval between subsequent EMRV actuation were the principle Oyster Creek specific features which resulted in demonstrating an increased robustness of the torus. Specifically, the OCPUA demonstrates that the Oyster Creek torus could tolerate a five valve initial simultaneous actuation followed by a five valve subsequent simultaneous actuation without taking credit for staggered setpoints or operator action. As such, staggered setpoints are not needed to demonstrate the structural integrity of the torus consistent with NUREG-0661 (Reference 6.9); and, represents margin in addition to that provided by the Long-Term Program reevaluation methodologies.

EMRV Actuation - Short Term Considerations (Cont'd.):

However, an assumption that the time between subsequent actuation be greater than 12 seconds was part of the EMRV load definition for the OCPUA. The turbine trip without bypass transient was re-analyzed with the current as set overpressure actuation and reset setpoints. That analysis was run without isolation condensers. The analysis results indicated that the interval would exceed 12 seconds by a substantial amount. The "as-left" setpoints of two (2) EMRVs at 1060 psig and three (3) EMRVs at 1080 psig, were used in the analysis above. These analyses would be required to be run each time these "as-left" actuation and/or reset setpoints are changed.

Other Considerations:

A detailed review of the resolution of EMRV issues associated with core and containment spray net positive suction head, torus temperature monitoring, systematic evaluation program, NUREG-0578, NUREG-0737, 10 CFR 50, Appendix R, 10 CFR 50.49, etc. was conducted to determine if the NRC's SER or equivalent GPUN resolutions were affected. The results of this review (Reference 6.1) indicated that none of these GPUN resolutions were sensitive to the high pressure actuation set or reset points.

4.0 NO SIGNIFICANT HAZARDS CONSIDERATIONS:

GPUN has determined that this Technical Specification Change Request involves no significant hazards consideration as defined by NRC in 10 CFR 50.92.

4.1 **Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident previously evaluated.**

The adequacy of the plant response to analyzed accidents is not sensitive to an increase in the EMRV overpressure actuation setpoint. Accidents analyzed each reload are the loss of coolant accident, fuel misloading and control rod drop accident. The overpressure actuation mode of the EMRVs does not contribute to mitigating the affects of these events.

With respect to the plant response to a small break loss of coolant accident, again, the overpressure actuation mode of the EMRVs does not contribute to mitigating the affects of these events.

The probability of the occurrence or the consequence of analyzed accidents are not sensitive to the EMRV overpressure action setpoint.

4.2 Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

With respect to the probability of experiencing a stuck open EMRV, which would represent an increase in the probability of a small break loss of coolant, the review of EMRV preventative maintenance results indicates that there are no overpressure actuation setpoint dependent phenomena which would increase the probability of the EMRV to stick open. The consequences of a stuck open EMRV are mitigated by the operational strategies provided by the emergency operating procedures; and, thus are not sensitive to the setpoint.

The analyses and evaluations performed indicate that a new or different kind of accident from any previously evaluated is not created. This is primarily because the overpressure actuation mode of the EMRVs is a preventative not a protective function with respect to fuel rod cladding and reactor coolant pressure boundary protection and the margins determined by the methodologies of the Mark I Containment Long-Term Program evaluation have not been reduced.

4.3 Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

There are three (3) dimensions to addressing this consideration. They are:

a. overpressurization events represent a challenge to fuel integrity as measured by the critical power ratio (CPR) projected to be expected during a transient or accident required to be analyzed as compared to the minimum critical power ratio (MCPR) required to be maintained;

b. overpressurization events represent a challenge to reactor coolant pressure boundary as compared to ASME code requirements associated with maintaining such a boundary; and,

c. EMRV(s) actuation represents a challenge to primary containment because such actuation(s) challenge the structural integrity of the EMRV branch connections to the Main Steam Line, EMRV discharge headers, "Y-quenchers", torus shell, torus attached piping and torus submerged structures.

The margins of safety associated with each of the above listed considerations are bounding value based. Therefore, as long as the plant response remains bounded by established quantitative values (e.g., ASME code allowables, minimum critical power ratio, etc) then the "margins of safety" have not been reduced.

The analyses performed utilizing the NRC approved GPUN reload analysis methodology, the reevaluation of the originally approved Mark I Containment Long-Term Program results and the comprehensive review of NRC's SER containing GPUN positions related to industry re-evaluations and upgrades indicates that the implicit or explicit margins of safety associated with the operation of the Oyster Creek Nuclear Generating Station will not be reduced.

The Commission has provided guidelines on the application of the three standards by listing specific examples in 45 FR 14870. The proposed amendment is considered to be in the same category as example (vi) of amendments "that are considered not likely to involve significant hazards consideration" in that, the proposed change may reduce a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to systems or components specified in the Standard Review Plan. As discussed above, the proposed amendment has been evaluated and demonstrated to represent no reduction in safety margin. Thus, operation of the facility in accordance with the proposed amendment involves no significant hazards considerations.

5.0 IMPLEMENTATION:

It is requested that the amendment authorizing this change become effective for Operating Cycle 15, i.e., at the restart from refueling outage 15R presently scheduled to start on or about September 10, 1994.

6.0 REFERENCES:

6.1 GPUN Topical Report, TR-101, "Considerations Associated with Changing EMRV Setpoints."

6.2 MPR[1] Report, MPR-550, "Oyster Creek Nuclear Generating Station Test Report - Effect of Modified Discharge Device on Response of Suppression Chamber to Relief Valve Actuation," May, 1978.

6.3 MPR Report, MPR-733, "Oyster Creek Nuclear generating Station Mark I Containment Long Term Program - Plant Unique Analysis Report Suppression Chamber and Vent System," August, 1982.

6.4 MPR Report, MPR-734, "Oyster Creek Nuclear generating Station Mark I Containment Long Term Program - Plant Unique Analysis Report Torus Attached Piping," August, 1982.

6.5 MPR Report, MPR-772, "Oyster Creek Nuclear generating Station Mark I Containment Long Term Program - Plant Unique Analysis Supplemental Report," July, 1983.

[1]. MPR Associates Inc.

6.0 REFERENCES Cont'd.:

6.6 MPR Report, MPR-999, "Oyster Creek Nuclear generating Station Mark I Containment Long Term Program - Addendum to MPR-734 Plant Unique Analysis Report," December, 1988.

6.7 MPR Report, MPR-1434, "Evaluation of Proposed Increase in Technical Specification Limits for EMRV Setpoint Pressure on Mark I Containment Long-Term Program Analyses," December, 1993.

6.8 NUREG-0484, "Methodology for Combining Dynamic Responses," September, 1978.

6.9 NUREG-0661, "Safety Evaluation Report - Mark I Containment Long-Term Program," July, 1980.