RISK ASSESSMENT OF DEFERRED

OYSTER CREEK PROJECTS

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1.0 INTRODUCTION

Provide Risk Analysis support of the planning activities that consider the following potential Oyster Creek scenarios:

1. Continued Operation to the end of licensed life (2009)
2. Sale of Oyster Creek to a third party
3. Early Shutdown in September, 2000

Specifically, provide the risk impact of the deferral of projects until the 18R refueling outage in support of the potential early shutdown in the year 2000.

2.0 METHOD

The Oyster Creek commitments have been reviewed and grouped into three categories by the Project and Regulatory Review Teams. The three categories are:

- Defer now, before final plant decision is made
- Cancel after final plant decision is made
- Implement as originally committed

For the projects which are to be deferred, provide a risk analysis of the impact of the deferral. In addition, provide an integrated assessment of the risk impact associate with the deferral of the proposed projects. The process for the evaluation is divided into four steps.

First, Evaluate the Status of the Projects within the framework of the various risk analysis studies performed for Oyster Creek. In this step it is determined whether the risk impact of project deferral can be reflected or inferred using the previously developed risk analysis studies.

- Review the available risk analysis studies (i.e., Probabilistic Risk Assessments (PRAs) and External Event (IPEEE) analyses.)
- Review the deferred projects.
- Define whether impact of the deferred projects can be directly or indirectly inferred from available risk evaluations.

Second, Evaluate the Safety or Risk Impact of the proposed deferred individual projects.

- If the risk impact of the deferral of the project can be directly produced using the available risk studies, perform the evaluation and provide the risk impact.
- If the risk impact cannot be directly inferred, however, minor modifications to existing evaluations can be performed, perform modifications and provide the risk impact.
- If the risk impact cannot be either directly or indirectly inferred from existing risk evaluations, either:
  - Perform additional risk evaluations and provide the impact, or

- Qualitatively assesses the risk impact in a framework that lends itself to incorporation with quantitatively assessed risk impacts produced in the steps above.

As part of this step, individual projects with significant risk impacts may be addressed in part. That is, risk significant portions of an individual project may be recommended for completion on the current schedule with the remainder of the project being deferred. If the risk impact is large and cannot be reduced by performing portions of the project or compensatory measures, the project will be recommended for completion on the original schedule. This ensures that the proposed risk increase remains small.

Third, Categorize All Safety/Risk Impacts using categories of high, medium and low. For the quantitatively produced risk impacts this consists of assigning numerical increases in core damage frequency or large early release frequency to pre-defined ranges. In the case of qualitatively evaluated impacts this consists of an assessment based on judgement. Assignment of a risk category allows for the integration of the risk impacts in cases where different figures of merit may be used to evaluate projects or activities.

Fourth, Evaluate the Integrated Safety/Risk Impact. Using the categories established in step three, provide a final integrated risk assessment. In the case of the qualitative evaluations, weighting factors based on judgement may be required. This step allows for the risk impacts to be considered in an integrated manner and as part of an overall risk management approach.

The figures of merit used in the evaluation of the quantitative risk impact are core damage frequency and large early release frequency (LERF). These figures of merit are chosen since most previously performed risk studies evaluate the frequency of core damage or large early release frequency. Other qualitative factors such as, consideration of alternative endstates, (e.g., significant transients) are documented in the individual evaluations. These qualitative factors can affect the allocation of a project to a given risk category.

In overview, the above methodology agrees closely with the methods for the use of PRA methods in risk informed decision making outlined in the NRC draft Standard Review Plan, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk Informed Decisionmaking: General Guidance" (reference 1). For comparison purposes:

- Steps 1 and 2 are equivalent to the first element in the draft SRP, Define the Proposed Change.
- Steps 2 through 4, correspond to Element 2 of the draft SRP, Conduct Engineering Evaluations.
- The third element of the draft SRP, Develop Implementation and Monitoring Strategies is also addressed in steps 2 through 4 on an individual project basis. Each project is evaluated for the potential for risk reduction, including compensatory measures. For example, fire watches have been posted in fire zones which contain thermolag fire barriers. The evaluation of Implementation and Monitoring strategies is performed on an activity or project basis depending on risk impact of the project deferral and the risk reduction achievable with potential compensatory measures. Also, performing parts or portions of projects are considered potential compensatory measures. For example, the most risk significant portions of a project may proceed as planned while less risk significant portions are deferred for a single cycle.
- The fourth element in the draft SRP is represented in the submittal of the integrated schedule to the NRC. The submittal and supporting documents contain sufficient information to support the conclusions of the acceptability of the deferrals and are available for staff review.

3.0 EVALUATION OF PROJECTS

As stated previously, the Oyster Creek commitments and projects have been reviewed and grouped into three categories by the Project and Regulatory Review Teams. The three categories are: Defer, Cancel or Implement as originally committed. The following projects have been proposed to be DEFERRED:

D.1. Generic Letter 96-06 Modifications
D.2. SQUG – Seismic Qualification Modifications
D.3. Control Room Human Factors Design Review (Back Panels)
D.4. Anticipatory Scram Logic Modification
D.5. Thermolag Fire Barrier Modifications
D.6. Severe Accident Management Guidelines
D.7. Reactor Water Cleanup Automatic Isolation Modification

Complete descriptions of the projects proposed for deferral is available in Appendix A of this report.

3.1 Step 1 – Evaluation of the Status of the Projects Proposed for Deferral

The goal of this step is to determine whether the risk impact of the deferral of the above projects can be estimated using the available risk analyses done for Oyster Creek. The risk analyses performed in support of Oyster Creek include the Oyster Creek Probabilistic Risk Assessment (OCPRA) and the Oyster Creek IPE for External Events (IPEEE). The Oyster Creek IPEEE includes a Seismic PRA as well as a Modified Fire PRA.

	Project or Activity	Applicable Risk Evaluation
D.1.	Generic Letter 96-06 Modifications	Level 2 OCPRA
D.2.	SQUG – Seismic Qualification Modifications	Seismic PRA
D.3.	Control Room Human Factor Design Review (Back Panels)	Qualitative
D.4.	Anticipatory Scram Logic Modification	OCPRA
D.5.	Thermolag Fire Barrier Modifications	Fire IPEEE
D.6.	Severe Accident Management Guidelines	Qualitative
D.7.	Reactor Water Cleanup Automatic Isolation Modification	Level 2 OCPRA

In the "Applicable Risk Evaluation" column the following are used: OCPRA, Level 2 OCPRA, Seismic PRA, Fire Individual Plant Examination for External Events (IPEEE), or Qualitative.

- The OCPRA (reference 2) refers to the plant specific Level 1 PRA performed in response to the IPE generic letter.
- The Level 2 OCPRA (reference 3) refers to the full scope Level 2 PRA performed in response to the IPE generic letter.
- The Seismic PRA and the Fire IPEEE refer to the quantitative evaluation performed in response to the IPE for External Events analysis (reference 4). The Oyster Creek Fire IPEEE is a modified probabilistic risk assessment due to the use of a screening approach. Details are available in reference 4 and Appendix C.
- In the case where no existing risk analysis can be used in the determination of the quantitative risk impact of the deferral of the project, then the term "Qualitative" is used in the "Applicable Risk Evaluation" column.

All evaluations performed to determine the risk impact of the deferral of projects are discussed in summary in the following report section and in detail in Appendix B.

3.2 Step 2 - Evaluate the Safety or Risk Impact

This report section provides a summary of the methods and results of the determination of the safety/risk impact of the deferral of projects. Details on the specific evaluations are available in Appendix B of this report.

3.2.1 Generic Letter 96-06 Modifications

Perform the proposed modifications in response to Generic Letter 96-06 during the 18R refueling outage. The generic letter questions the operability of systems with regard to their capability to withstand ambient heating following a loss of coolant accident (LOCA). Preliminary analysis indicates that two of the three issues contained in the generic letter do not apply to Oyster Creek (reference 6). The third issue, containment penetration overpressurization due to ambient heating following isolation during a LOCA applies to Oyster Creek. Without overpressure protection, the concern is that entrapped water between the inboard and outboard isolation valves is heated, expands, and increases in pressure challenging the strength of the particular penetration.

Operability determinations have been performed indicating that all systems considered susceptible to overpressure are considered operable for the interim duration until either procedural changes and or hardware modifications can be made (reference 6,7,8). GPU has committed to perform corrective actions which involve physical modifications to the plant be documented in the Integrated Schedule for Oyster Creek, pursuant to license condition 2.C.(6) of the Full Term Operating License.

The analysis of the risk impact of the deferral of the 96-06 modifications until the 18R outage is performed using insights developed in the Level 1 and Level 2 OCPRAs. LOCAs which discharge to the drywell and result in core damage are adjusted to reflect endstates which bypass the primary containment. Three sensitivity cases are evaluated to determine the risk impact of project deferral. Case 1, evaluates the risk impact if all LOCAs which discharge to the drywell airspace are assumed to fail the containment integrity. Case 2, evaluates the risk impact if large LOCAs which discharge to the drywell airspace are assumed to fail the containment integrity. Case 3, evaluates the risk impact if 50% of large LOCAs which discharge to the drywell airspace are assumed to fail the containment integrity.

Based on this analysis, the large early release frequency increase for case 3, is 5.5×10^{-8} per year. This equates to a 7.4% in the large early release frequency. Details of the analysis and results are presented in Appendix B.

3.2.2 Seismic Qualification Modifications – Phase II

The scope of the project is to implement modifications which address outliers resulting from Oyster Creek's unresolved safety issue (USI) A-46 Program which was performed in response to the NRC's Generic Letter 87-02. Seismic verification walkdowns performed utilizing SQUG methodology were conducted during 1994 (reference 24). Phase I modifications have been completed.

The evaluation for the risk impact of this modification is performed using insights from the Seismic PRA performed in support of Generic Letter 88-20, Supplement 4 (reference 4). Modifications on the core spray and containment spray anchorage and the platform in the southwest corner room are expected to be completed on schedule. Other SQUG modifications, with the exception of the diesel generator building roof slabs, were evaluated in the Seismic PRA fragility analysis "as-built" and therefore, do not significantly affect risk.

Three sensitivity cases are evaluated with respect to the capacity of the diesel generator building roof. The first, models the capacity of the roof at 0.18g which provides a 50% chance of building failure given the safe shutdown earthquake (SSE). The second and third cases, model a capacity of the diesel generator building roof at 0.36g and 0.54g, which correspond to 2 and 3 times the SSE acceleration. These cases are based on the fact that seismically designed equipment typically has a capacity of 2 to 3 times design (i.e., SSE).

The results for case 1, 2 and 3 are core damage frequency increases of 2.2×10^{-6}, 1.0×10^{-6} and 3.6×10^{-7} per year, respectively. This corresponds to a 61.1%, 28.6% and 9.9% increase in the Seismic PRA core damage frequency, respectively. Details of the analysis and results are presented in Appendix B.

3.2.3 Control Room Human Factors Design Review

In summary, the work included within the scope of this project includes the upgrade of the human engineering of the control room back panels 1R through 5R and 6R through 11R (including 9XR) as well as 11XR, 12R, 12XR, 14R, 14XR ,11F and 16R (reference 23). The scope of work for each of the panels includes:

1. Review of the panels by GPU Human Factors
2. Walkdowns with Plant Operations
3. Development of three sets of drawings of these panels (relabeling, label specifications, and final "as-builts".
4. Repainting, relabeling, and annuciator matched demarcation of these panels, including necessary cosmetic panel repairs (e.g., sanding, hole filling, etc.)

To date, many plant changes have resulted in improved back panels. Since the initial control room human factors review, significant changes to the back panels have occurred. Control room panels have been upgraded or replaced as a result of many recent plant modifications. Panel equipment is replaced or upgraded in accordance with current company standards which meet or exceed those of NUREG-0700. Major modifications include:

1. Main Generator Protection Upgrade Project. This project affected panels 11R, 11XR, 12R and 12XR.
2. Digital Feedwater and Digital Recirculation Control Modification. This project affected panels 8R and 9R.
3. Recirculation Flow Scram Electronics Modification. This modification affected panels 3R and 5R.
4. Panel 2R was relabeled in accordance with the Back Panel Labeling Project.

In addition to the above mentioned projects, other plant programs, initiatives and corrective actions have identified poorly or confusingly labeled equipment on the control room panels which have since been relabeled in accordance with company standards.

No quantitative figure of merit is available for the assessment of the risk impact of the deferral of this modification. The current plant specific PRAs are based on the current control room design. Modifications

to the back panels in the control room may increase the human/machine interface, however it is likely that this will not significantly affect the probabilities of operator errors.

3.2.4 Anticipatory Scram Bypass Logic Improvement

This project was initiated to improve upon actions taken in response to LER 95-005 (reference 12). The actions taken to date include the reseting of PSH switches to conservative setpoints. This conservatism is required, to assure that under certain plant configurations, where steam is redirected, that thermal power remains below 40% when these anticipatory SCRAM signals are bypassed. Because the PSH switches tap off the third stage extraction steam lines, the operation of the switches is not a true indicator of reactor thermal power. They are a better indicator of turbine load. Thus the parameters monitored by the PSH switches are not indicative of total plant thermal power except during normal "full power" plant steam alignments.

The modification would replace the current PSH switches with more precise switches. In addition, local control switches and indicating lamps would be installed to provide indication when the PSH switches are closed. The new PSH switches will provide a permissive signal that will allow bypassing of the affected anticipatory SCRAM signals. Group annunciation of when the anticipatory scram bypass is permitted will be provided to the control room. This will allow return of the setpoints from the current 25% to the 40% power level.

Currently, operators are not aware when the turbine stop valve closure and turbine control valve fast closure scrams are bypassed (i.e., no control room or local indication). Lack of indication of when the scrams are bypassed results in lost generation due to unnecessarily low power reductions when turbine scrams must be bypassed (e.g., grid work). In addition, without indication of the engaged scram signal bypass, operators could assume that the scram is engaged when in fact it is not, resulting in an inadvertent scram.

The risk of deferring this project from the 17R to the 18R refueling outage is estimated using the insights and results of the Level 1 OCPRA. Since, not performing the modification in the 17R refueling outage could result in the potential for an inadvertent scram (reference 11) and the safety significance of the non-conservative setpoint is considered minimal (reference 12), the turbine trip frequency is increased by one turbine trip over the operating cycle. Details to the risk evaluation are contained in Appendix B.

3.2.5 Thermo-Lag Fire Barrier Modifications

The scope of this project is to install modifications to bring the Thermo-Lag 330-1 fire barrier systems installed at Oyster Creek into compliance with 10CFR50 Appendix R. The NRC has raised several concerns as to adequacy of these systems and has issued information notice 92-46, Bulletins 92-01, and 92-01, Supplement 1 and Generic Letter 92-08 on this subject. The NRC now considers the fire rating of these systems to be indeterminate and is requiring compensatory measures (i.e., fire watches) until the issue is resolved.

The risk impact of altering the current completion date of the Thermo-Lag project is estimated using the Oyster Creek Individual Plant Examination for External Events (reference 4). Specifically, the risk impact is estimated using the Fire Individual Plant Examination (IPEEE).

The core damage frequency estimates produced in the Oyster Creek Fire IPEEE do not model the effect of the Thermo-Lag fire barriers therefore, upgrade of the Thermo-Lag barriers would serve to reduce the current estimates of the core damage frequency. However, due to the high combustible loading as well as the importance of the 480 VAC system in the mitigation of fire events at Oyster Creek the modifications of the 480 VAC Switchgear Rooms are scheduled to proceed as planned. Details of the evaluation are presented in Appendix B.

3.2.6 Severe Accident Management Guideline Development

Development and implementation of the Severe Accident Management Guidelines (SAMGs) is currently scheduled to be completed by December 1998. This submittal presents the risk impact of the deferral of development and implementation of the SAMGs until December of 2000. The risk impact of this deferral is estimated qualitatively since actions directed by the SAMGs are not modeled in any existing risk evaluations and have not been completely developed. The guidance for coping with severe accidents is expected to provide limited risk benefit over the two year period remaining between the completion of the guidance and the potential closure date of Oyster Creek (Fall 2000). On this basis, the deferral of the implementation of the Severe Accident Management Guideline is assigned to the low risk category.

3.2.7 Reactor Water Cleanup Leakage Monitoring, LOCA Detection and Isolation

The purpose of these modifications is two-fold. First, the installation of thermocouples on the discharge of relief valves in the reactor water cleanup (RWCU) system to allow for the easy determination of leakage by operations or maintenance (reference 17). Second, the installation of temperature sensors at the entrance to the reactor water cleanup recirculating pump room to detect leaks which constitute a LOCA on the high pressure portions of the RWCU system (reference 18).

The installation of thermocouples on the discharge of the relief valves in the RWCU system to allow easy determination of leakage by operation or maintenance is being performed for "improved radiological conditions" and dose reduction. The risk impact of the deferral of this modification is considered low based on judgement and the fact that this modification was initially proposed for the purposes of does reduction and convenience.

The installation of temperature sensors to detect leaks in the RWCU which constitute a LOCA on the high pressure portions of the RWCU system is being performed in response to the long term corrective actions for GE Nuclear SIL 604 (reference 19) and Oyster Creek Deviation Report 96-1097 (reference 20). The concern of the GE SIL is that at certain power levels below 100%, automatic isolation of the cleanup system on low reactor water level may not occur due to the capacity of the feedwater system to maintain level despite inventory losses out the break.

The risk impact is determined using insights from the Level 1 and Level 2 OCPRAs. The risk impact is determined by defining an new initiating event representing a break in the RWCU line. The initiating event frequency is based on the probability of the RWCU pipe break including the failure to isolate probability (i.e., operator response) to the event. The initiating event impact conservatively assumes the failure of the all equipment in the reactor building due to the harsh environment following the failure to isolate the break. In this fashion the increase in core damage frequency is estimated. This initiating event results in the bypass of the primary containment. Therefore, the large early release frequency is also estimated using insights from the Level 2 OCPRA.

The increase in core damage frequency due to a RWCU line break in the high pressure sections is 1.2×10^{-7} per year. This equates to a 3.2% increase. The large early release frequency is calculated to be 1.2×10^{-7} per year, the same as the core damage frequency increase. The percent increase in the large early release frequency is 16%. Details of the analysis and results are presented in Appendix B.

3.3 Step 3 - Categorize Safety/Risk Impacts

This report section provides the categorization of the safety/risk impact for each of the proposed project deferments. The risk impacts are categorized into either high, medium or low categories according to the increase in total core damage or large early release frequency as defined on Tables 1 and 2, below. In the case where the risk impact categorization differs between the core damage frequency and the large early release frequencies, the higher of the two categorizations is assigned.

In the case where non-quantitative results have been used to assess the risk impact, the assignment of the risk category is based on judgement. For the quantitative assessments, additional details on the assignment of the risk category are provided in the detailed analysis, Appendix B. For the non-quantitative assessment, details are provided in this report section.

Table 1 – Categorization of Risk Impacts Affecting Core Damage Frequency

Risk Category	Core Damage Frequency Percent Increase Range
High – High	1000%
High	100% - 1000%
Medium	10% - 100%
Low	<10%

Table 2 – Categorization of Large Early Release Frequency Increases

Risk Category	Range (Percent Increase)
High	> 100%
Medium	10% – 100%
Low	<10%

Table 1, Categorization of Risk Impacts Affecting Core Damage Frequency, is derived from the Oyster Creek On-Line Maintenance Risk Management Procedure (reference 16) and, in part, from the EPRI PRA Applications Guide (reference 15). Table 2 – Categorization of Large Early Release Frequency Increases, is derived, in part, from the EPRI PRA Applications Guide which indicates that increases in large early release frequency for Oyster Creek of greater than 36.4% are significant and require additional analysis. Using the criteria in Tables 1 and 2 above, the risk impact is given in Table 3 and discussed in the following paragraphs.

Generic Letter 96-06 Modifications. The Generic Letter 96-06 Modifications do not affect the core damage frequency since the concern is the overpressure of piping penetrations following loss of coolant accidents. This overpressure is conservatively assumed to result in a failure of the piping penetration such that the primary containment integrity is compromised. If this assumption is applied to all losses of coolant the increase in the large early release frequency is 34.3%. Small LOCAs which discharge to the drywell may not result in the same environment (i.e., slower containment and piping heatup) versus large LOCAs. If small LOCAs which discharge to the drywell are excluded, the increase in the large early release frequency increase becomes 14.8%. However, if the probability of GL 96-06 related pipe break given a large LOCA is not unity (1.0) then the risk impact would be less. The same is true if the probability of containment integrity failure is not unity (1.0) given a LOCA (e.g., specific break location(s)). Assuming a GL 96-06 pipe break occurs in only 50% of the LOCA cases, the risk increase is less than 10% in large early release frequency. Using Table 2, the increase is then categorized as Low.

SQUG – Seismic Qualification Modifications. The SQUG phase I modifications have been completed. Significant phase II modifications are recommended for completion, including the core spray and containment spray pump anchorages and the platform in the southwest corner room. The remaining modifications, with the exception of the diesel generator building roof slabs, were included in the Seismic PRA as "as-built". Therefore, the current Seismic PRA includes the capacity of these components. The

diesel generator building roof slab anchorage was not evaluated "as-built". Reducing the capacity of the building results in an increase in the seismic core damage frequency.

To evaluate the risk impact of the diesel generator building roof, three cases are evaluated. The first case, reduces the capacity of the diesel generator building to 0.18 g reflecting a 50% chance of building failure given the safe shutdown earthquake (SSE). The second and third cases reflect higher capacities of 0.36g and 0.54g, which is equal to the 2 and 3 times the SSE, respectively. The cases are based on the compensatory measure of verifying the diesel generator building roof anchorage for the SSE. Typically, seismically designed equipment has a mean fragility of 2 to 3 times the SSE acceleration. The core damage frequency estimates in case 3 indicate a Low category. Case 2 has an increase in core damage frequency equal to $1x10^{-6}$ which is traditionally considered low.

Control Room Human Factors Design Review (Back Panels). The control room human factors design review for the back panels does not significantly affect either the core damage or large early release frequencies. Many changes have been completed since the proposal of the project. The remaining modifications are not expected to significantly improve the human/machine interface. As such, the deferral of this project is assigned to the Low category based on judgement.

Anticipatory Scram Logic Modification. The risk impact of the anticipatory scram logic modification is assessed using the Level 1 OCPRA. Deferring the modification is assumed to result in a turbine trip over the operating cycle for which it is deferred. This results in an increase in the core damage frequency of 6.8%. The impact on the large early release frequency is not calculated since turbine trip events typically result in intact containment endstates. Using Table 1, the risk impact of the deferral of this modification is assigned to the Low category.

Thermo-Lag Fire Barrier Modifications. The risk impact of the deferral of the Thermo-Lag modifications is estimated using the Fire IPEEE. The Fire IPEEE did not model the effect of the Thermo-Lag fire barriers. As such, upgrade of the barriers would serve to lower the existing core damage frequency estimates. All fire zones which contain Thermo-Lag were screened in the fire analysis (i.e., core damage frequency less than $1x10^{-6}$ per year) with the single exception of the "A" 480 VAC Switchgear Room. Given the importance of the 480 VAC Switchgear to the mitigation of transients at Oyster Creek, it is recommended that the Thermo-Lag replacement for both "A" and "B" 480 VAC Switchgear Rooms proceed as originally planned. The remaining fire zones containing Thermo-Lag were screened in the fire analysis (i.e., core damage frequency less than $1x10^{-6}$ per year). Therefore, these fire zones are considered less risk significant. On this basis, the risk impact of the deferral of the Thermo-Lag Upgrade, excluding the 480 VAC Switchgear Rooms, is categorized as Low.

Severe Accident Management Guidelines. The deferral of the Severe Accident Management Guidelines does not significantly affect the core damage frequency or large early release frequency. The PRAs reflect the plant design, maintenance and operations practices at the time of their development and do not include the affect of the guidelines. The Severe Accident Management Guidelines may provide limited benefit over the two year period between the completion of the guidance and the potential closure date of Oyster Creek (Fall 2000). On this basis, the deferral of the implementation of the Severe Accident Management Guideline is assigned to the Low risk category.

Reactor Water Cleanup Automatic Isolation Modification. The risk impact of the deferral of the Reactor Water Cleanup Automatic Isolation Modification is estimated using the Level 1 and Level 2 OCPRAs. Estimation of the frequency and impacts of a break in the high pressure portion of the RWCU piping indicate that the large early release frequency increases considerably. Given the significant increase in the large early release frequency as well as the potential for a bypass of the primary containment, it is recommended that this modification be implemented as originally planned. The risk impact of this

modification is assigned to the high category based on the increase in the large early release frequency and degradation to a significant fission product boundary.

Table 3 – Risk Impact Categories of Project Deferrals

Project Title	CDF Increase	LERF Increase	Percentage Increase	Risk Category
Generic Letter 96-06 Modifications [1]	None	5.5×10^{-8}	7.4%	Low
SQUG – Seismic Qualification Modifications	3.6×10^{-7} [2]	N/A	9.9%	Low [2]
Control Room Human Factors Design Review (Back Panels) [3]	Small	N/A	N/A	Low
Anticipatory Scram Logic Modification	2.6×10^{-7}	N/A	6.8%	Low
Thermo-Lag Fire Barrier Modifications [3], [4]	Small [4]	N/A	N/A	Low [4]
Severe Accident Management Guidelines [3]	Small	Small	N/A	Low
Reactor Water Cleanup Automatic Isolation Modification – Case 1 (Case 2 in brackets) [6]	1.2×10^{-7} (8.0×10^{-7})	1.2×10^{-7} (8.0×10^{-7})	16% [5] (106%)	Medium (High [5])

1. Uses the results from sensitivity case 3 which are deemed to best represent the risk impact.
2. Case 3 and a risk category of Low is displayed. A low category is assigned due to compensatory measures.
3. Qualitatively assessed.
4. Excludes the 480 VAC Switchgear Rooms which are to be performed as scheduled.
5. Percentage increase is in terms of the large early release frequency.
6. Recommended to be complete on schedule due to degradation of significant fission product barrier and relatively high LERF.

3.4 Step 4 - Evaluate the Integrated Safety/Risk Impact

The evaluation of the integrated risk impact is performed in two steps. The first step involves the determination of whether the six projects for deferral are independent. That is, does the risk impact of the deferral of the projects have dependencies which influence the overall risk impact in a non-linear fashion. For example, if two or more projects were to affect a fission product barrier, a single project may have a low risk impact while the combined affect of the two or more projects could have a significant or high risk impact. Dependencies are uncovered through the review of the projects to determine if the deferral of the project:

- Impacts the same system, structure or component (SSC)
- Affects the same safety function
- Affects the same fission product boundary or
- Reduces the margin of safety for multiple accidents (e.g., external, internal or shutdown events)

A review of the detailed risk evaluations indicate that there are no dependency issues. In the absence of any risk impact dependencies, the risk impact can be calculated using simple addition of quantitative risk impacts. Since the evaluation contains qualitative assessments as well as conservative quantitative results and, different figures of merit (i.e., core damage frequency and large early release frequency), judgement is used in the combination of the integrated risk impacts.

The total core damage frequency increase is 6.2×10^{-7} per year. The total large early release frequency is 5.5×10^{-8} per year. On this basis the integrated risk assessment would be considered low and therefore acceptable. The projects which were evaluated non-quantitatively have small risk impacts either in the core damage or large early release frequency. In addition, the projects which are performed on the current schedule, either in total or in part, can present reductions in the total core damage frequency or large early release when compared with existing risk studies. Quantitative values have not been developed for these

risk reductions, however they can significantly offset any risk increase. Therefore, the integrated assessment of the total risk impact of the deferred projects is considered low.

4.0 CONCLUSIONS

This analysis evaluated the risk impact of the deferral of projects for a single cycle. The risk impact was evaluated using, to the extent possible, the existing plant specific PRAs and risk evaluations. Where the risk impact could not be evaluated using the existing risk studies, minor changes were made to allow their use, or the risk impact was performed qualitatively. To assess the total risk impact of the deferral of the projects, individual as well as integrated risk impact evaluations were undertaken.

In addition to the risk impact evaluation, attention was paid to defense in depth and maintaining adequate safety margins as well as ensuring that the incremental change in risk was small. To this end, projects with significant risk impact were recommended for implementation on the original schedule. Portions or elements of projects which contributed significantly to the risk impact or which represented significant decreases in the safety margin or defense in depth were also recommended for completion as originally scheduled. Performing several projects on schedule, either in total or in part, can reduce risk when compared with existing risk evaluations and serve to offset, in part, the small risk increases.

Of the seven (7) projects originally scheduled for deferral only four (4) are deferred in their entirety. These are:

- Generic Letter 96-06 Modifications
- Control Room Human Factors Design Review (Back Panels)
- Anticipatory Scram Logic Modification
- Severe Accident Management Guidelines.

Of the remaining three (3) projects, one (1) is recommended for implementation as originally scheduled (Reactor Water Cleanup Automatic Isolation Modification) and two (2) are only deferred in part (SQUG – Seismic Qualification Modifications and Thermo-Lag Fire Barrier Modifications). With the implementation as planned of the projects and portions of projects which have significant risk impact, the individual risk impact as well as the integrated risk impact remains acceptable.

5.0 REFERENCES

1. Nuclear Regulatory Commission, Standard Review Plan, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk Informed Decisionmaking: General Guidance", Draft for Comment, Draft SRP Chapter 19, Revision L, March 27, 1997.

2. GPU Nuclear Corporation, "Oyster Creek Probabilistic Risk Assessment (Level 1)", Volumes 1 through 6, November 1991.

3. GPU Nuclear Corporation, "Oyster Creek Probabilistic Risk Assessment (Level 2)", Volume 1, June 1992.

4. GPU Nuclear Corporation, "Oyster Creek Individual Plant Examination for External Events", December 1995.

5. Nuclear Regulatory Commission, "Generic Letter 96-006: Assurance of Equipment Operability & Containment Integrity During Design Basis Accident Conditions", September 30, 1996.

6. GPU Nuclear Corporation, Letter 6730-97-2012, "Generic Letter No. 96-06, Dated September 30, 1996, 120-day Response", January 28, 1997.

7. GPU Nuclear Corporation, Letter 6730-97-2059, "Generic Letter No. 96-06, Dated September 30, 1996, 120-day Response, Updated", February 26, 1997.

8. GPU Nuclear Corporation, Letter 6730-97-2145, "Generic Letter No. 96-06, Dated September 30, 1996, Corrective Actions Follow-up to 120-day Response Letter", June 3, 1997.

9. GPU Nuclear Corporation, Technical Data Report, "Oyster Creek Applicability to Generic Letter 96-06", TDR-1210, Revision 1, March 18, 1997.

10. GPU Nuclear Corporation, Request for Project Approval, "Seismic Qualification Modifications – Phase II", Budget Activity Number 403092, Revision 0, August 7, 1997.

11. GPU Nuclear Corporation, Request for Project Approval, "Anticipatory Scram Bypass Logic Improvement", BA Number 400018, Revision 0, August 7, 1997.

12. GPU Nuclear Corporation, Licensee Event Report, Oyster Creek Unit 1, "Anticipatory Scram Bypass Switches' Non-Conservative Setpoint Due to Original Plant Design", LER 95-005, Revision 0, August 25, 1995.

13. GPU Nuclear Corporation, Request for Project Approval, "Thermo-L[illegible] Fire Barrier Modification", BA Number 403042, Revision 1, August [illegible], 1997.

14. GPU Nuclear Corporation, Fire Protection Program, "Fire H[illegible] Analysis Report", Document Number 990-1746, Revision 9, Volumes 1 and 2.

15. Electric Power Research Institute (EPRI), "Probabilistic Safety Assessment Applications Guide", EPRI TR-105396, Final Report, August 1995.

16. GPU Nuclear Corporation, Oyster Creek Policy and Procedure Manual, "On-Line Maintenance Risk Management", 2000-ADM-3022.01, Revision 4, September 5, 1997.

17. GPU Nuclear Corporation, Request for Project Approval, "Cleanup System Leakage Monitoring", Activity Number 40G294, Revision 0, August 7, 1997.

18. GPU Nuclear Corporation, Request for Project Approval, "RWCU LOCA Detect & Isolate (SIL604)", Budget Activity Number 400017, Revision 0, August 7, 1997.

19. GE Nuclear Energy, Service Information Letter (SIL), "Reactor Water Clean-up System Break Detection", SIL No. 604, November 6, 1996.

20. GPU Nuclear Corporation, Deviation Report, "Deviation Report of RWCU Break Detection", DVR Number 96-1097, December 4, 1996.

21. PLG, Incorporated, "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants (Failure Data)", PLG-0500, Volume 2, Revision 0, July 1989.

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23. GPU Nuclear Corporation, Request for Project Approval, "Control Room Human Factors Design Review", Budget Activity Number 328030, Revision 2, August 7, 1997.

24. GPU Nuclear Corporation, Request for Project Approval, "Seismic Qualification Modifications – Phase II", Budget Activity Number 403092, Revision 0, August 7, 1997.

25. GPU Nuclear Corporation, GPU Nuclear Testing Procedure, "Control Room and Cable Spreading Room Heatup Test", Procedure 254/13-MTX 26.12.2.6.

26. GPU Nuclear Corporation, GPU Nuclear Calculation, "PRA Control Room Loss of Ventilation Profile", Calculation C-1302-862-5360-010, Revision 0.

APPENDIX A

DEFFERRED PROJECT LIST

NRC Regulatory Commitments
Defer Before Final Plant Decision

Commitment	Regulatory Approach
Generic Letter 96-06 Modifications – Pressure concerns for piping penetrations (BA #31G690, BA #320011)	Integrated Schedule Update
SQUG – Seismic Qualification Modifications. NOTE: Significant number of modifications have been completed. (BA #403092)	Integrated Schedule Update
Control Room Human Factors Design Review – Repaint, refurbish, and relabel control room panel 1R through 10R, 6XR, 11XR, 12R, 12XR, 14R, 14XR, 16R, 11F, 9XR and 11P. NUREG 0737, Supplement 1 (BA #328030)	Integrated Schedule Update
Anticipatory Scram Logic Modification LER 95-05 (BA #400018)	Integrated Schedule Update
Severe Accident Management Program Generic Letter 88-20 – "Individual Plant Examination for Severe Accident Vulnerabilities"	Integrated Schedule Update
ThermoLag Fire Barrier Modifications 16 and 17R. NOTE: Modifications to 460V rooms will not be deferred. (BA #403042)	Integrated Schedule Update
Reactor Water Clean Up – Provide an automatic RWCU system isolation on a line break – SIL 604 – LER 96-015. (BA #40G294, BA #400017)	Integrated Schedule Update

APPENDIX B

RISK IMPACT ANALYSIS

This appendix provides details on the risk impact analysis done in support of the project deferral. Only the projects evaluated for quantitative analysis are described here.

B.1 Generic Letter 96-06 Modifications

The generic letter questions the operability of systems with regard to their capability to withstand ambient heating following a loss of coolant accident (LOCA). Preliminary analysis indicates that two of the three issues contained in the generic letter do not apply to Oyster Creek (reference 6). The third issue has been determined to apply to Oyster Creek.

Project Description and Proposed Change

The third issue is the overpressurization of containment penetrations due to ambient heating following isolation during a LOCA. Without overpressure protection, the concern is that entrapped water between the inboard and outboard isolation valves is heated, expands, and increases in pressure challenging the strength of the particular penetration. Five (5) penetrations require modification to relieve overpressure:

1. Reactor Building Closed Cooling Water (RBCCW) Return from the Drywell
2. Shutdown Cooling Supply to the Reactor
3. Isolation Condenser "A" - Condensate Return
4. Isolation Condenser "B" - Condensate Return
5. Recirculation Sampling

Operability determinations have been performed indicating that all systems considered susceptible to overpressure are operable for the interim duration until either procedural changes and/or hardware modifications can be made (reference 6,7,8). GPU has committed to perform corrective actions which involve physical modifications to the plant be documented in the Integrated Schedule for Oyster Creek, pursuant to license condition 2.C.(6) of the Full Term Operating License.

Risk Impact Evaluation

The analysis of the risk impact of the deferral of the 96-06 modifications until the 18R outage is performed using insights developed in the Level 1 and Level 2 OCPRAs. The Generic Letter 96-06 is primarily concerned with the integrity of the containment following a LOCA. That is, the overpressure of containment penetrations resulting in failure of the penetration and containment integrity. The figure of merit or risk measure used in the determination of the risk impact of the 96-06 modifications is, therefore, Large Early Release Frequency. Three cases are used in the estimate of the risk impact. The cases are ordered from most conservative to least conservative.

Sensitivity Case 1

In Case 1, the risk impact is estimated by assuming that all LOCAs which discharge to the drywell **and** result in core damage, also result in overpressurization and failure of a containment penetration. This includes the effect of small LOCAs even though small LOCAs would not result in the severe environmental conditions that occur during large LOCAs. The contribution of all LOCAs to the total core damage frequency is taken from the Level 1 OCPRA. Table B-1 provides the contribution of all LOCAs to the total core damage frequency.

Since it is assumed that piping overpressure results in the failure of a containment penetration, a large containment bypass is created. This is a conservative assumption since a large containment bypass requires either of the following to occur: (1) a single large pipe rupture at the containment penetration or (2) two pipe breaks with one inside the containment and another outside.

The contribution of the LOCAs is normally an "containment intact" plant damage endstate. To model the assumed containment integrity failure, the normal plant damage endstate of "containment intact" is adjusted to large early release endstate. This leads to increase in the total large early release frequency approximately equal to the core damage frequency of the LOCA contributions. A "Large Early Release Frequency Worksheet" (LERF) is provided as Table B-2 and displays the estimation of the increase in LERF.

From Table B-1, the frequency of all LOCAs with discharge to the primary containment airspace is equal to 2.59×10^{-7} per year. In Table B-2, the Level 1 Key Plant Damage State, PIFW, which is a "containment intact" endstate, is reduced by the above LOCA frequency of 2.59×10^{-7} per year.

Key Plant Damage State PIFW (Base Case) – LOCA Frequency Contribution = New PIFW KPDS

$$1.16 \times 10^{-6} - 2.59 \times 10^{-7} = 8.98 \times 10^{-7}$$

The percent variance on Table B-2, is then 8.98×10^{-7} divided by the base case of 1.16×10^{-6} or -22%.

Also on Table B-2, the Level 1 Key Plant Damage State, MKCU, which is a large early release containment endstate, is increased by the above LOCA frequency of 2.59×10^{-7} per year.

Key Plant Damage State MKCU (base case) + LOCA Frequency Contribution = New MKCU KPDS

$$1.72 \times 10^{-7} + 2.59 \times 10^{-7} = 4.31 \times 10^{-7}$$

The percent variance on Table B-2, is then 4.31×10^{-7} divided by the base case of 1.72×10^{-7} or +151%.

The changes to these key plant damage states results in an increase of the large early release frequency from the base case of 7.56×10^{-7} per year to 1.02×10^{-6} per year or 2.59×10^{-7} per year. This corresponds to an increase in the large early release frequency of 34.3%.

The Case 1 analysis of the risk impact of the deferral of the Generic Letter 96-06 modifications remains bounding due to the conservative assumptions regarding pipe rupture, pipe rupture locations as well as the assumption that small LOCAs result in the overpressurization of the susceptible containment penetrations.

Sensitivity Case 2

Case 2, a less conservative sensitivity case, is evaluated to estimate a less conservative risk impact. This sensitivity case evaluates the risk impact assuming that the issue of piping overpressure is restricted to the large LOCAs into the drywell airspace which result in core damage. That is, small LOCAs result in a less severe environment due to the slower heatup of the drywell. The slower heatup allows for the initiation of containment spray and/or the automatic depressurization system. The effect of the cooling of the containment spray system and use of the automatic depressurization system to remove heat to the torus and results in less heat being discharged to the drywell. With less ambient heatup of the drywell and, therefore less ambient heatup of piping penetrations, it is less likely that piping failures due to overpressurization will occur. Using the "Large LOCA with discharge to the drywell airspace" row from Table B-1, the evaluation performed in case 1 above is repeated. The results are displayed on Table B-4.

In the sensitivity case, the increase in large early release frequency is 1.12×10^{-7} per year which corresponds to a 14.8% increase. This evaluation remains conservative due to assumptions with regard to assumed pipe breaks following exceeding code allowable stresses and the assumed break location (or multiple breaks) which fail containment integrity.

Sensitivity Case 3

In case 3 the risk impact is evaluated by assuming that piping overpressure is restricted to the large LOCAs into the drywell airspace which result in core damage **and** only 10% of piping overpressurizations result in

a pipe break which fails containment isolation. This is reasonable assuming that ultimate failure pressures of pipes are typicaily significantly higher than the design or code allowable pressures. For the total of five penetrations this is equal to 5 times 10%, or a 50% chance of containment integrity failure due to pipe overpressurization.

The effect of small LOCAs is also not considered in this case. As stated in the evaluation of case 2, small LOCAs result in a less severe environment due to the slower heatup of the drywell. The slower heatup allows for the initiation of containment spray and/or the automatic depressurization system. The cooling effect of the containment spray system and use of the automatic depressurization system to remove heat to the torus, results in less heat being discharged to the drywell. With less ambient heatup of the drywell and piping penetrations it is less likely that piping failures due to overpressurization will occur. The frequency used is 50% of the frequency in case 2.

In this sensitivity case, the increase in large early release frequency is $5.5x10^{-8}$ per year which corresponds to a 7.4% increase.

Results and Conclusions

The results of three sensitivity cases used to evaluate the affect of the deferral of Generic Letter 96-06 Modifications is displayed on Table B.5, below. As the results indicate, the large early release frequency ranges from a percent increase of 7.4% to 34.3%.

Increases in the Large Early Release Frequency are categorized according to the criteria on Table 2 (found in the main report). The risk impact of the sensitivity cases range over risk categories of Low and Medium. Based on judgement, case 3 is deemed to best represent the deferral of Generic Letter 96-06 Modifications and the risk impact is categorized as low.

Table B.5 – Summary of Generic Letter 96-06 Risk Evaluations

Case Description	Large Early Release Frequency		Risk Category
	Increase Value	Percent Increase	
Case 1: All LOCA core damage frequency contributions (which discharge to drywell airspace) result in containment integrity failure.	$2.59x10^{-7}$	34.3%	Medium
Case 2: All Large LOCA core damage frequency contributions (which discharge to the drywell airspace) result in containment integrity failure.	$1.12x10^{-7}$	14.8%	Medium
Case 3: 50% of Large LOCA core damage frequency contributions (which discharge to the drywell airspace) result in containment integrity failure.	$5.5x10^{-8}$	7.4%	Low

TABLE B-1 OCPRA INITIATING EVENT IMPORTANCE

MODEL Name: OCPRA-13
Initiator Contributions to End State Group : ALL
Total Frequency for the Group = 3.7982E-06

Initiator	Frequency	Unaccounted	Percent
LOSP	1.24E-06	1.00E-09	32.73%
TTRIP	4.64E-07	3.64E-09	12.23%
RT	2.84E-07	1.80E-09	7.48%
LOFW	2.60E-07	1.40E-09	6.85%
CMSIV	2.57E-07	3.18E-09	6.76%
LOTB	1.48E-07	9.61E-10	3.90%
LOCV	1.48E-07	2.69E-09	3.89%
LOIS	1.22E-07	7.26E-10	3.21%
EPRL	1.19E-07	2.54E-09	3.13%
LBI	1.09E-07	8.30E-11	2.87%
LOFC	1.02E-07	2.69E-09	2.69%
SBI	9.46E-08	2.01E-10	2.49%
IEMRV	9.04E-08	1.64E-09	2.38%
PLOFW	7.83E-08	1.89E-09	2.06%
LBIO	7.65E-08	2.86E-11	2.01%
SAI	5.24E-08	2.13E-10	1.38%
SBO	4.57E-08	2.25E-11	1.20%
LOIA	3.15E-08	1.96E-09	0.83%
EPRH	2.93E-08	1.52E-09	0.77%
LOCW	2.15E-08	1.35E-09	0.57%
IADS	1.59E-08	9.01E-11	0.42%
SAOTB	2.28E-09	4.89E-11	0.06%
LAICS	1.48E-09	5.36E-11	0.04%
LAIMS	1.37E-09	5.30E-11	0.04%
LAOMS	4.62E-10	3.36E-12	0.01%
LACIC	3.99E-11	3.36E-12	0.00%
SAORB	2.66E-11	1.03E-11	0.00%
SAOIC	8.52E-12	1.24E-11	0.00%
TOTALS	3.80E-06	2.98E-08	100.00%
Large LOCAs with discharge to the drywell airspace	**1.12E-07**	**1.90E-10**	**2.94%**
All LOCAs with discharge to the drywell airspace	**2.59E-07**	**6.04E-10**	**6.81%**

Table B-2: 96-06 LERF Estimation Spreadsheet

Reference Case:	Base Case (Risk Model: OCPRA-1.3)
Case under study:	96-06 LERF (Case 1)

Level 1 - Initiating Events

I.E.	Value	Reference	Variance
CMSIV	4.17E-01	4.17E-01	0%
EPRH	5.61E-02	5.61E-02	0%
EPRL	1.76E-01	1.76E-01	0%
IADS	1.33E-03	1.33E-03	0%
IEMRV	3.31E-02	3.31E-02	0%
LAICS	8.21E-05	8.21E-05	0%
LAIMS	1.15E-04	1.15E-04	0%
LAOIC	6.96E-08	6.96E-08	0%
LAOMS	6.44E-08	6.44E-08	0%
LBI	5.67E-04	5.67E-04	0%
LBIO	8.37E-06	8.37E-06	0%
LOCV	2.24E-01	2.24E-01	0%
LOCW	2.71E-02	2.71E-02	0%
LOFC	1.71E-01	1.71E-01	0%
LOFW	1.51E-01	1.51E-01	0%
LOIA	4.33E-02	4.33E-02	0%
LOIS	7.51E-03	7.51E-03	0%
LOSP	3.26E-02	3.26E-02	0%
LOTB	1.03E-02	1.03E-02	0%
PLOFW	1.78E-01	1.78E-01	0%
RT	7.21E-01	7.21E-01	0%
SAI	9.27E-03	9.27E-03	0%
SAOIC	1.59E-06	1.59E-06	0%
SAORB	7.70E-07	7.70E-07	0%
SAOTB	3.64E-04	3.64E-04	0%
SBI	7.81E-03	7.81E-03	0%
SBO	2.86E-06	2.86E-06	0%
TTRIP	8.97E-01	8.97E-01	0%

Level 1 - Key Plant Damage States

Input KPDS	Input Value	Reference ***Base Case***	Percent Variance
PIFW	8.98E-07	1.16E-06	-22%
NIFW	1.04E-06	1.04E-06	0%
OIAU	5.75E-07	5.75E-07	0%
OJAU	1.83E-07	1.83E-07	0%
MKCU	4.31E-07	1.72E-07	151%
MJAU	5.88E-08	5.88E-08	0%
NJHW	1.56E-08	1.56E-08	0%
Total CDF	3.80E-06	3.80E-06	0%

LERF Estimation

	Percent of CDF Analyzed =		84.36%
	Total analyzed frequency =		3.20E-06
Category 1A - Large Early			
MKCU	100%	4.31E-07	4.31E-07
NIFW	30.85%	1.04E-06	3.22E-07
OIAU	0.95%	5.75E-07	5.47E-09
		Total	7.58E-07
	Percent of Total Analyzed =		23.65%
Category 1B - Containment Bypass			
OJAU	100%	1.83E-07	1.83E-07
MJAU	100%	5.88E-08	5.88E-08
NJHW	100%	1.56E-08	1.56E-08
		Total	2.57E-07
	Percent of Total Analyzed =		8.03%
Total LERF (sum of above) =			1.02E-06
	Reference LERF =		7.56E-07
	Percent Change in LERF =		**34.26%**
	Percent of Total Analyzed =		31.69%

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Risk significant cutoffs:	Risk Significant Cutoff	Delta for this Case
CDF	51.40%	0.00%
LERF	36.37%	34.26%

Comments:

Table B-3: 96-06 LERF Estimation Spreadsheet

Reference Case:	Base Case (Risk Model: OCPRA-13)
Case under study:	96-06 LERF (Case 2)

Level 1 - Initiating Events

I.E.	Value	Reference	Variance
CMSIV	4.17E-01	4.17E-01	0%
EPRH	5.61E-02	5.61E-02	0%
EPRL	1.76E-01	1.76E-01	0%
IADS	1.33E-03	1.33E-03	0%
IEMRV	3.31E-02	3.31E-02	0%
LAICS	8.21E-05	8.21E-05	0%
LAIMS	1.15E-04	1.15E-04	0%
LAOIC	6.96E-08	6.96E-08	0%
LAOMS	6.44E-08	6.44E-08	0%
LBI	5.67E-04	5.67E-04	0%
LBIO	8.37E-06	8.37E-06	0%
LOCV	2.24E-01	2.24E-01	0%
LOCW	2.71E-02	2.71E-02	0%
LOFC	1.71E-01	1.71E-01	0%
LOFW	1.51E-01	1.51E-01	0%
LOIA	4.33E-02	4.33E-02	0%
LOIS	7.51E-03	7.51E-03	0%
LOSP	3.26E-02	3.26E-02	0%
LOTB	1.03E-02	1.03E-02	0%
PLOFW	1.78E-01	1.78E-01	0%
RT	7.21E-01	7.21E-01	0%
SAI	9.27E-03	9.27E-03	0%
SAOIC	1.59E-06	1.59E-06	0%
SAORB	7.70E-07	7.70E-07	0%
SAOTB	3.64E-04	3.64E-04	0%
SBI	7.81E-03	7.81E-03	0%
SBO	2.86E-06	2.86E-06	0%
TTRIP	8.97E-01	8.97E-01	0%

Level 1 - Key Plant Damage States

Input KPDS	Input Value	Reference ***Base Case***	Percent Variance
PIFW	1.04E-06	1.16E-06	-10%
NIFW	1.04E-06	1.04E-06	0%
OIAU	5.75E-07	5.75E-07	0%
OJAU	1.83E-07	1.83E-07	0%
MKCU	2.84E-07	1.72E-07	65%
MJAU	5.88E-08	5.88E-08	0%
NJHW	1.56E-08	1.56E-08	0%
Total CDF	3.80E-06	3.80E-06	0%

LERF Estimation

	Percent of CDF Analyzed =		84.35%
	Total analyzed frequency =		3.20E-06
Category 1A - Large Early			
MKCU	100%	2.84E-07	2.84E-07
NIFW	30.85%	1.04E-06	3.22E-07
OIAU	0.95%	5.75E-07	5.47E-09
		Total	6.11E-07
	Percent of Total Analyzed =		19.07%
Category 1B - Containment Bypass			
OJAU	100%	1.83E-07	1.83E-07
MJAU	100%	5.88E-08	5.88E-08
NJHW	100%	1.56E-08	1.56E-08
		Total	2.57E-07
	Percent of Total Analyzed =		8.03%
Total LERF (sum of above) =			8.68E-07
	Reference LERF =		7.56E-07
	Percent Change in LERF =		**14.82%**
	Percent of Total Analyzed =		27.10%

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Risk significant cutoffs:	Risk Significant Cutoff	Delta for this Case
CDF	51.40%	0.00%
LERF	36.37%	**14.82%**

Comments:

Table B-4: 96-06 LERF Estimation Spreadsheet

Reference Case:	Base Case (Risk Model: OCPRA-13)
Case under study:	96-06 LERF (Case 3)

Level 1 - Initiating Events

I.E.	Value	Reference	Variance
CMSIV	4.17E-01	4.17E-01	0%
EPRH	5.61E-02	5.61E-02	0%
EPRL	1.76E-01	1.76E-01	0%
IADS	1.33E-03	1.33E-03	0%
IEMRV	3.31E-02	3.31E-02	0%
LAICS	8.21E-05	8.21E-05	0%
LAIMS	1.15E-04	1.15E-04	0%
LAOIC	6.96E-08	6.96E-08	0%
LAOMS	6.44E-08	6.44E-08	0%
LBI	5.67E-04	5.67E-04	0%
LBIO	8.37E-06	8.37E-06	0%
LOCV	2.24E-01	2.24E-01	0%
LOCW	2.71E-02	2.71E-02	0%
LOFC	1.71E-01	1.71E-01	0%
LOFW	1.51E-01	1.51E-01	0%
LOIA	4.33E-02	4.33E-02	0%
LOIS	7.51E-03	7.51E-03	0%
LOSP	3.26E-02	3.26E-02	0%
LOTB	1.03E-02	1.03E-02	0%
PLOFW	1.78E-01	1.78E-01	0%
RT	7.21E-01	7.21E-01	0%
SAI	9.27E-03	9.27E-03	0%
SAOIC	1.59E-06	1.59E-06	0%
SAORB	7.70E-07	7.70E-07	0%
SAOTB	3.64E-04	3.64E-04	0%
SBI	7.81E-03	7.81E-03	0%
SBO	2.86E-06	2.86E-06	0%
TTRIP	8.97E-01	8.97E-01	0%

Level 1 - Key Plant Damage States

Input KPDS	Input Value	Reference *Base Case*	Percent Variance
PIFW	1.10E-06	1.16E-06	-5%
NIFW	1.04E-06	1.04E-06	0%
OIAU	5.75E-07	5.75E-07	0%
OJAU	1.83E-07	1.83E-07	0%
MKCU	2.28E-07	1.72E-07	33%
MJAU	5.88E-08	5.88E-08	0%
NJHW	1.56E-08	1.56E-08	0%
Total CDF	3.80E-06	3.80E-06	0%

LERF Estimation

	Percent of CDF Analyzed =		84.35%
	Total analyzed frequency =		3.20E-06
Category 1A - Large Early			
MKCU	100%	2.28E-07	2.28E-07
NIFW	30.85%	1.04E-06	3.22E-07
OIAU	0.95%	5.75E-07	5.47E-09
		Total	5.55E-07
	Percent of Total Analyzed =		17.32%
Category 1B - Containment Bypass			
OJAU	100%	1.83E-07	1.83E-07
MJAU	100%	5.88E-08	5.88E-08
NJHW	100%	1.56E-08	1.56E-08
		Total	2.57E-07
	Percent of Total Analyzed =		8.03%
Total LERF (sum of above) =			8.12E-07
	Reference LERF =		7.56E-07
	Percent Change in LERF =		**7.41%**
	Percent of Total Analyzed =		25.35%

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Risk significant cutoffs:	Risk Significant Cutoff	Delta for this Case
CDF	51.40%	0.00%
LERF	36.37%	7.41%

Comments:

B.2 Seismic Qualification Modifications – Phase II

The scope of the project is to implement modifications which address outliers resulting from Oyster Creek's unresolved safety issue (USI) A-46 Program which was performed in response to the NRC's Generic Letter 87-02. Seismic verification walkdowns performed utilizing SQUG methodology were conducted during 1994 (reference 24). Phase I modifications have been completed.

Project Description and Change

The specific work scope for this project includes modification to the following:

1. Anchorage of the Core Spray Main Pumps
2. Anchorage of the Core Spray Booster Pump (P-20-002A)
3. Anchorage of the Containment Spray Pumps (P-21-001B, P-21-001C and P-21-001D)
4. Anchorage for panel ER-661-100 (Turbine Building Ragems Panel) which is a missile hazard to two Safe Shutdown Equipment List (SSEL) components.
5. Modify anchorage for MCC 1A12 and 1B12. MCC 1A12 is an SSEL component and MCC 1B12 is an interaction hazard for cabling for MCC 1A12.
6. Modifications for the CRD Hydraulic Control Units. The existing installed anchorage can be made adequate for the majority of the units provided their stiffness is improved. The individual unit which stands alone (HCU-305-34-51) will require a more significant modification to provide additional anchorage.
7. Provide new anchorage for the MSIV solenoid rack in the drywell.
8. The platform supporting T-22-001 (reactor building equipment drain tank) in the southwest corner room is seismically inadequate. A modification is required to ensure that the core spray pumps and associated cabling is not jeopardized by platform failure.
9. Replacement of relays in the diesel generator control circuits, the rotary inverter control cabinet, ASCO Transfer Switches.
10. Modification of the 4160 VAC Switchgear circuitry.
11. Modification of the anchorage for the diesel generator roof slabs.

Currently this work is scheduled for completion by the end of 1998. Consider the deferral of work until 18R refueling outage.

Risk Impact Evaluation

The evaluation for the risk impact of this modification is performed using insights from the Seismic PRA performed in support of Generic Letter 88-20, Supplement 4.

Due to the timing of the IPEEE project and the seismic qualification of equipment (SQUG) project, fragility calculations performed in support of the IPEEE relied, to varying degrees, on the SQUG work packages. The varying degree of reliance on SQUG packages produces fragility calculations which range from those based on the SQUG packages alone (including modifications credited in the SQUG packages, if any) to those based on the seismic IPEEE fragility walkdowns. Where no SQUG package existed when seismic fragility walkdowns were performed, the plant equipment fragility was based on the "as-built" condition. Where a completed SQUG package existed, the fragility value was based on the SQUG package. In this case the fragility would be based on any planned modifications, if the equipment was a SQUG outlier. If the equipment passed the SQUG evaluation, the SQUG package would be used in the development of the fragility.

The use of the SQUG packages in the fragility evaluation results in several fragilities, which are based on the successful completion of the SQUG modifications. Such is the case with items 1, 2, 3 and 8 from the list of planned modifications. Given this fact, as well as the importance of the core spray and containment

spray systems in the mitigation of transient events at Oyster Creek, it is recommended that items 1, 2, 3 and 8 be implemented as planned.

SQUG modification items 4 and 5 are included in the SQUG list to provide support for control room ventilation including the requirement for power supplies for recovery of ventilation using portable fans. Ventilation studies done is support of the OCPRA, including actual test data, indicate that control room ventilation is not required for a significant period of time following its loss (reference 25). Therefore, control room ventilation was not required for success in the OCPRA or the Seismic PRA. On this basis, the risk impact of these SQUG modification is judged to less significant.

SQUG modification items 6 through 10 (excluding item 8) were included in the Seismic PRA fragility analysis. Fragility analysis of these items was based on the "as-built" plant during the seismic fragility walkdowns. As such, these SQUG modifications are expected to have a limited affect on the risk associated with seismic events. On this basis, the deferral of these modification items is considered low.

The final item in the SQUG modification list is the modification of the diesel generator building roof slabs. This modification is required due to corrosion of the existing anchorage. The fragility analysis did not explicitly include the roof anchorage of the diesel generator building roof slabs. Currently, the diesel generator building is considered operable. Periodic verification of the condition of the diesel generator building roof slab anchorage is performed.

In order to estimate the risk impact of the deferral of this modification, the capacity of diesel generators themselves (assuming that building failure during a seismic event fails both diesel generators) is adjusted. Three case studies are performed.

In the first case (case 1), a fragility value (i.e., capacity) of 0.18g mean acceleration is used. This estimate provides a 50% probability that the diesel generator building fails during a safe shutdown earthquake (SSE). Two additional case studies are performed assuming that the diesel generator building roof higher capacity. In case 2, the diesel generator building roof is assumed to have a capacity of 2 times the SSE or 0.36g. In case 3, the diesel generator building roof is assumed to have a capacity of 3 times the SSE or 0.54g. These values are chosen since seismically designed systems or structures typically have capacities 2 to 3 times the design capacity. This assertion is logical since at the design acceleration equipment is likely to be successful and the capacity or fragility referred to in this document represent the mean failure accelerations. Table B-6, provides a description and the results of the evaluation.

Results and Conclusions

The estimation of the risk impact of the Seismic Qualification Modifications – Phase II assumes that due to the importance of the core spray and containment spray systems in the mitigation of transients at Oyster Creek, these modifications (items 1, 2, 3 and 8) proceed as originally scheduled. In addition, it is assumed that modifications which affect the control room ventilation (items 4 and 5) and those for which fragility evaluation were performed on the "as-built" plant (items 6 through 10, excluding item 8) are of low risk significance.

The core damage frequency increase ranges from 61.1% to 9.9% using assumed capacities of 0.18 to 0.54g for the diesel generator building roof slabs. Provided that the capacity of the diesel generator building roof is verified to meet design, then case 3 (0.54g capacity) is judged to best represent deferral of the Seismic Qualification Modifications. Case 3 is categorized as low based on the percent increase in core damage frequency. It should be noted that the increase in core damage frequency in case 2 is 1.0×10^{-6} per year which is typically considered low.

Current compensatory measures include the periodic verification of the condition of the diesel generator building roof slab anchorage. A verification (i.e., testing) of the capacity of the anchorage should be performed to provide less conservative estimates of the true capacity of the anchors.

Table B-6
Summary of Seismic Qualification Modification (Phase II) Risk Impacts

Case Description	Seismic Acceleration (g)	EDG Roof Failure Probability	Core Damage	
			Frequency (increase)	Percent Increase
Case 1: Diesel Generator Building Roof Slabs Capacity Equal to 0.18g.	0.007 – 0.26	1.62×10^{-2}	5.8×10^{-6} (2.2×10^{-6})	61.1%
	0.26 – 0.46	9.17×10^{-1}		
	0.46 – 0.62	9.97×10^{-1}		
	0.62 – 0.82	1.00		
Case 2: Diesel Generator Building Roof Slabs Capacity Equal to 0.36g.	0.007 – 0.26	1.23×10^{-3}	4.6×10^{-6} (1.0×10^{-6})	28.6%
	0.26 – 0.46	4.13×10^{-1}		
	0.46 – 0.62	8.74×10^{-1}		
	0.62 – 0.82	9.83×10^{-1}		
Case 3: Diesel Generator Building Roof Slabs Capacity Equal to 0.54g.	0.007 – 0.26	1.36×10^{-4}	4.0×10^{-6} (3.6×10^{-7})	9.9%
	0.26 – 0.46	1.43×10^{-1}		
	0.46 – 0.62	6.28×10^{-1}		
	0.62 – 0.82	9.08×10^{-1}		

B.3 Anticipatory Scram Bypass Logic Improvement

This project was initiated to improve upon actions taken in response to LER 95-005. The actions taken to date include the reset PSH switches to conservative setpoints. This conservatism is required, with the existing plant configuration, to assure that under certain plant configurations, where steam is redirected, that thermal power remains below 40% when these anticipatory SCRAM signals are bypassed. Because the PSH switches tap off the third stage extraction steam lines, the operation of the switches is not a true indicator of reactor thermal power. The PSH switches are a better indicator of turbine load. Thus the parameters monitored by the PSH switches are not indicative of total plat thermal power except during normal "full power" plant steam alignments.

Project Description and Proposed Change

The modification would replace the current PSH switches with more precise switches, with hysteresis sufficient to lesson contact bouncing and with narrower dead bands. In addition, auxiliary relays would be installed with local control switches and indicating lamps at the turbine standard to provide indication when the PSH switches are closed. The new PSH switches will provide a permissive signal that will allow bypassing of the affected anticipatory SCRAM signals. Group annunciation of when the anticipatory scram bypass is permitted will be provided to the control room using existing spare wires to the control room. (Additional wiring will be required). This will allow return of the setpoints from the current 25% to the 40% power level.

Currently, operators are not aware when the turbine stop valve closure and turbine control valve fast closure scrams are bypassed (i.e., no control room or local indication). Lack of indication of when the scrams are bypassed results in lost generation due to unnecessarily low power reductions when turbine scrams must be bypassed (e.g., grid work). In addition, without indication of the engaged scram signal bypass, operators could assume the scram is engaged when it is fact is not, resulting in an inadvertent scram.

Risk Impact Evaluation

The risk of deferring this project from the 17R to the 18R refueling outage is estimated using the insights and results of the Level 1 OCPRA. Since, not performing the modification in the 17R refueling outage could result in the potential for an inadvertent scram (reference 11) and the safety significance of the non-conservative setpoint is considered minimal (reference 12), the turbine trip frequency is increased by one turbine trip over the operating cycle.

Table B-7, Oyster Creek Initiating Event Contribution, provides the initiating event contributions to total core damage frequency. The turbine trip initiating event frequency of 0.897 per year is adjusted to reflect the potential for an additional turbine trip due to the deferral of the anticipatory scram modification for one cycle. That is, the turbine trip frequency is increase by an additional turbine trip each cycle (1 / two years) or by 0.5 per year for a new turbine trip frequency of 1.397 per year. The results of this model (risk model: TTRIP) are provided on Table B-8, TTRIP Model Initiating Event Contributions.

Results and Conclusion

The total core damage frequency increases from 3.80×10^{-6} per year to 4.06×10^{-6} per year or 6.8%. The turbine trip initiating event (TTRIP) increases in contribution from a 12% contributor in the base case to an 18% contributor in the TTRIP risk model. The risk impact is categorized according to the ranges specified on Table 2 (main report section). These ranges are those used in the Risk Management of On-Line Maintenance Program at Oyster Creek (reference 16). Since the increase in the total core damage frequency is 6.8%, the risk category is Low.

TABLE B-7: OCPRA INITIATING EVENT CONTRIBUTION

MODEL Name: OCPRA-13 Initiator Contributions to End State Group : ALL Total Frequency for the Group = 3.7982E-06			
Initiator	Frequency	Unaccounted	Percent
LOSP	1.24E-06	1.00E-09	32.73%
TTRIP	4.64E-07	3.64E-09	12.23%
RT	2.84E-07	1.80E-09	7.48%
LOFW	2.60E-07	1.40E-09	6.85%
CMSIV	2.57E-07	3.18E-09	6.76%
LOTB	1.48E-07	9.61E-10	3.90%
LOCV	1.48E-07	2.69E-09	3.89%
LOIS	1.22E-07	7.26E-10	3.21%
EPRI	1.19E-07	2.54E-09	3.13%
LBI	1.09E-07	8.30E-11	2.87%
LOFC	1.02E-07	2.69E-09	2.69%
SBI	9.46E-08	2.01E-10	2.49%
IEMRV	9.04E-08	1.64E-09	2.38%
PLOFW	7.83E-08	1.89E-09	2.06%
LBIO	7.65E-08	2.86E-11	2.01%
SAI	5.24E-08	2.13E-10	1.38%
SBO	4.57E-08	2.25E-11	1.20%
LOIA	3.15E-08	1.96E-09	0.83%
EPRH	2.93E-08	1.52E-09	0.77%
LOCW	2.15E-08	1.35E-09	0.57%
IADS	1.59E-08	9.01E-11	0.42%
SAOTB	2.28E-09	4.89E-11	0.06%
LAICS	1.48E-09	5.36E-11	0.04%
LAIMS	1.37E-09	5.30E-11	0.04%
LAOMS	4.62E-10	3.36E-12	0.01%
LAOIC	3.99E-11	3.36E-12	0.00%
SAORB	2.66E-11	1.03E-11	0.00%
SAOIC	8.52E-12	1.24E-11	0.00%
TOTALS	3.80E-06	2.98E-08	100.00%

TABLE B-8: TTRIP MODEL INITIATING EVENT CONTRIBUTION

Model Name: TTRIP Initiator Contributions to End State Group: ALL Total Frequency = 4.0572E-06			
Initiator	Frequency	Unaccounted	Percent
LOSP	1.24E-06	1.00E-09	30.64%
TTRIP	7.24E-07	4.12E-09	17.83%
RT	2.84E-07	1.80E-09	7.00%
LOFW	2.60E-07	1.40E-09	6.42%
CMSIV	2.57E-07	3.18E-09	6.33%
LOTB	1.48E-07	9.61E-10	3.65%
LOCV	1.48E-07	2.69E-09	3.65%
LOIS	1.22E-07	7.26E-10	3.00%
EPRL	1.19E-07	2.54E-09	2.93%
LBI	1.09E-07	8.30E-11	2.68%
LOFC	1.02E-07	2.69E-09	2.52%
SBI	9.46E-08	2.01E-10	2.33%
IEMRV	9.04E-08	1.64E-09	2.23%
PLOFW	7.83E-08	1.89E-09	1.93%
LBIO	7.65E-08	2.86E-11	1.89%
SAI	5.24E-08	2.13E-10	1.29%
SBO	4.57E-08	2.25E-11	1.13%
LOIA	3.15E-08	1.96E-09	0.78%
EPRH	2.93E-08	1.52E-09	0.72%
LOCW	2.15E-08	1.35E-09	0.53%
IADS	1.59E-08	9.01E-11	0.39%
SAOTB	2.28E-09	4.89E-11	0.06%
LAICS	1.48E-09	5.36E-11	0.04%
LAIMS	1.37E-09	5.30E-11	0.03%
LAOMS	4.62E-10	3.36E-12	0.01%
LAOIC	3.99E-11	3.36E-12	0.00%
SAORB	2.66E-11	1.03E-11	0.00%
SAOIC	8.52E-12	1.24E-11	0.00%
TOTALS	**4.06E-06**	**3.03E-08**	**100%**

B.4 Thermo-Lag Fire Barrier Modifications

The scope of this project was to install modifications to bring the Thermo-Lag 330-1 fire barrier systems installed at Oyster Creek into compliance with 10CFR50 Appendix R.

Project Description and Proposed Change

The fire barriers will be upgraded by overlaying the existing Thermo-Lag 330-1 with fire barrier material from another vendor. If the plant configuration does not have sufficient space for the additional material on a specific fire barrier, the Thermo-Lag will be removed and replaced with new fire barrier material. If any power cable cannot accept the additional capacity de-rating from the application of additional material, the Thermo-Lag will be removed and replaced with new fire barrier material or the cable rerouted to achieve an acceptable configuration.

The NRC has raised several concerns as to adequacy of these systems and has issued information notice 92-46, Bulletins 92-01, and 92-01, Supplement 1 and Generic Letter 92-08 on this subject. The NRC now considers the fire rating of these systems to be indeterminate and is requiring compensatory measures (i.e., fire watches) until the issue is resolved.

Risk Impact Evaluation

The risk impact of altering the current completion date of the Thermo-Lag project is estimated using the Oyster Creek Individual Plant Examination for External Events (reference 4). Specifically, the risk impact is estimated using the Fire Individual Plant Examination (IPEEE).

The Oyster Creek Fire IPEEE methodology is a modified PRA methodology which uses an iterative screening approach to remove from detailed evaluation (screen) those plant fire areas and zones which present low risk (i.e., less than $1x10^{-6}$ per year core damage frequency). More detailed analysis is then performed on those fire areas and zones which do not screen (i.e., core damage frequency greater than $1x10^{-6}$ per year). As stated above the approach is iterative in nature and involves three steps:

1. The first step involves the "all engulfing fire" in which the Fire IPEEE models the failure of all equipment within and cables which transit a given fire zone. In addition, all failure modes are addressed including "hot shorts" and a conservative transient model is chosen (e.g., all EMRVs open for pressure relief which requires all EMRVs to reclose).

2. For those fire zones whose core damage frequency does not screen (i.e., is not less than $1x10^{-6}$ per year) a second iteration is performed. In the second iteration, "*revised core damage frequency estimate*", the assumptions used in the development of the risk model as well as simple recoveries (such as ventilation restoration) are credited. Fire suppression probabilities, both manual and automatic are modeled only in more detailed evaluations.

3. For those fire zones whose core damage frequency does not screen (i.e., is not less than $1x10^{-6}$ per year) in the second iteration, a third and final iteration is performed. In the third iteration, the "*detailed core damage frequency estimate*", automatic fire suppression probabilities are modeled as well as factors concerning fire growth and propagation.

Using the above approach to the quantification of the core damage frequency due to fire events results in two fire zones which do not screen. These fire zones are the Cable Spreading Room[1] and the "A" 480 VAC Switchgear Room. The core damage frequency produced by the initial and revised estimates of core damage frequency (i.e., the first and second iterations) remain upper bound estimates since detailed evaluation is not performed for these areas. It is likely that a detailed evaluation would result in lower estimations of the core damage frequency for the screened fire zones. As such, it is not appropriate to make judgements with regards to core damage frequency or the risk importance ranking of an individual fire zone which was screened without addressing the conservative assumptions made in the analysis. That is, comparisons between screened fire zones with core damage frequencies of, for example, $7x10^{-7}$ per year and $3x10^{-7}$ may not be valid without investigating the various assumptions performed in the analysis.

Thermo-Lag protection of circuits was not modeled in the Fire IPEEE. That is, circuits protected by Thermo-Lag were assumed to fail due to the fire event in all iterations performed to evaluate the core damage frequency. A more complete overview of the methodology used in the development of the Fire IPEEE is presented in Appendix C.

Seven (7) fire zones at Oyster Creek use Thermo-Lag to provide a fire barrier. These fire areas are presented in Table B.9 below.

Table B.9 – Summary of Oyster Creek Fire Zones Containing Thermo-Lag

Fire Area/Zone		Combustible Loading		Core Damage Frequency
Designator	Description	BTUs / Sq. Ft.	Rating	
OB-FZ-06A	Office Bldg "A" 480 VAC Swgr Room	176601	High	5.1E-6
OB-FZ-06B	Office Bldg "B" 480 VAC Swgr Room	142101	High	3.1E-7 [a]
RB-FZ-01D	Reactor Building - 51' Elevation	20362	Low	2.7E-7 [c]
RB-FZ-01E	Reactor Building - 23' Elevation	24717	Low	1.3E-7 [c]
RB-FZ-01F2	Reactor Building - (-19') Elevation	964	Low	9.0E-7 [b]
TB-FZ-11C	Turbine Bldg, Swgr Rm, West End	13575	Low	4.6E-7 [b]
TB-FZ-11D	Turbine Building - Basement South End	35163	Low	2.1E-7 [c]

Notes:
(a) – Fire zone screened in initial evaluation assuming "all engulfing fire".
(b) – Fire zone screened in a revised evaluation including refined risk modeling.
(c) – Fire zone was screened following detailed evaluation including application of fire severity factor, fire detection and suppression.

Results and Conclusion

The core damage frequency estimates produced in the Oyster Creek Fire IPEEE do not model the effect of the Thermo-Lag fire barriers therefore, upgrade of the Thermo-Lag barriers would serve to reduce the current estimates of the core damage frequency. However, due to the high combustible loading as well as the importance of the 480 VAC system in the mitigation of fire events at Oyster Creek, the modifications of the 480 VAC Switchgear Rooms are scheduled to proceed as planned.

Based on the fact that modeling of the Thermo-Lag fire barriers could result in lower core damage frequency estimations, the risk impact of the deferral of the project, with the exclusion of the 480 VAC Switchgear Rooms, is assigned a low category.

[1] The cable spreading room did not screen in the detailed evaluation. This fact is provided for completeness. The cable spreading room does not contain circuits protected by Thermo-Lag.

B.5 Reactor Water Cleanup Leakage Monitoring, LOCA Detection and Isolation

The purpose of these modifications is two-fold. First, the installation of thermocouples on the discharge of relief valves in the reactor water cleanup (RWCU) system to allow for the easy determination of leakage by operations or maintenance (reference 17). Second, the installation of temperature sensors at the entrance to the reactor water cleanup recirculating pump room to detect leaks which constitute a LOCA on the high pressure portions of the RWCU system (reference 18).

Project Description and Proposed Change

The installation of thermocouples on the discharge of the relief valves in the RWCU system to allow easy determination of leakage by operation or maintenance is being performed for "improved radiological conditions" and dose reduction. The risk impact of the deferral of this modifications is considered low based on judgement and the fact that this modification was initially proposed for the purposes of dose reduction and convenience.

The installation of temperature sensors to detect leaks in the RWCU which constitute a LOCA on the high pressure portions of the RWCU system is being performed in response to the long term corrective actions for GE Nuclear SIL 604 (reference 19) and Oyster Creek Deviation Report 96-1097 (reference 20). The concern of the GE SIL is that at certain power levels below 100%, automatic isolation of the cleanup system on low reactor water level may not occur due to the capacity of the feedwater system to maintain level despite inventory losses out the break. Operator action to isolate the break is assumed not to occur for 10 minutes in licensing analysis. Thus, the mass release may be greater than previously analyzed. Concerns on the affect of additional mass release on the environmental qualification of equipment as well as radiological consequences have arisen. The scope of this modification is to install temperature sensors outside the Reactor Water Cleanup Room. These temperature sensors would be used to generate an isolation signal upon indication of a LOCA that would isolate V-16-1, V-16-2, V-16-14 and V-16-61.

Actions taken in response to the deviation report include: EQ Evaluations for Potentially Affected Safety Related Components, Additional Operator Guidance (Alarm Response Procedures), Additional Operator Training, and a Safety Evaluation. Planned actions include the design and implementation Automatic Isolation Modification.

Risk Impact Evaluation

The risk impact of the deferral of the proposed modification is determined using insights from the Level 1 and Level 2 OCPRAs. The Level 1 OCPRA models a large number of loss of coolant including those outside the primary containment. However, a large loss of coolant from below the reactor core and outside the containment (in the high pressure section of the RWCU system) was not originally modeled. The low pressure section of the RWCU system was modeled in the Interfacing Systems LOCA (ISLOCA) analysis (Appendix B.3 of the Level 1 OCPRA). The failure of the low pressure section of the RWCU system was thought to be dominant and therefore breaks in the high pressure sections were not addressed. The estimate of the risk impact of a break in the high pressure piping of the RWCU, therefore, requires an additional initiating event.

The frequency of this initiating event is determined on Table B.10. Generic data for the failure of piping sections (reference 21) is multiplying the number of hours in a year and by the approximate number of piping sections between the primary containment and the pressure control valve.

Pipe Break Probability (in sections/hour)	*	No. of hours/year	*	No. of Pipe Sections	=	RWCU LOCA Annual Frequency

The calculation of the initiating event frequency also includes the failure to isolate probability based on operator response to the event and the failure of the motor operated isolation valves to close on demand. The individual motor operated valve failure as well as the common mode failures of the valves is modeled. Generic data is used for both the individual valve failure as well as the common cause failure probabilities (reference 22). The result of the addition of the operator error probability and the mechanical failure of the motor operated valves is the failure probability to isolate the RWCU line break.

Operator Isolates RWCU Break	+	MOVs Fail to Close on Demand (including common causes)	=	Failure to Isolate RWCU Pipe Break

The failure to isolate combined (multiplied) with pipe break frequency results in the initiating event for the Unisolated RWCU line break.

RWCU LOCA Annual Frequency	*	Failure to Isolate RWCU Pipe Break	=	Unisolated RWCU Line Break

The initiating event impact conservatively assumes the failure of the all equipment in the reactor building due to the harsh environment following the failure to isolate the break.[2] This assumption is conservative since short term early operation of the core spray system will most likely occur. The feedwater system injects into the downcomer of the reactor vessel and since the break is below the reactor core, it will exit through the break without providing core cooling. With the failure of the core spray system (failed as a result of the initiating event impact), core damage is assumed to occur. (With the short term operation of the core spray system, parallel injection valves will be open, allowing injection of the fire protection system using manual valves located outside the reactor building wall.)

It is not necessary to exercise the Level 2 OCPRA since with a large break below the reactor core feedwater cannot provide sufficient cooling inventory. That is, the feedwater system injects into the downcomer of the reactor vessel and will exit the break without providing core cooling. With the failure of the core spray system (failed as a result of the initiating event impact), core damage is assumed to occur.

Since this initiating event also results in the bypass of the primary containment, the large early release fraction is estimated. Insights from the Level 2 OCPRA are used to develop Table B.11, RWCU Line Break LERF Estimation Spreadsheet.

In the estimation of the large early release frequency increase, the initiating event frequency (which is equal to the core damage frequency increase) is assigned to a containment bypass endstate (i.e., designator: OJAU). The increase in large early release is therefore equal to the initiating event frequency and core damage frequency increase.

The increase in core damage frequency due to RWCU line break of the high pressure sections is 8.0×10^{-7} per year. This equates to a 21.1% increase. The large early release frequency is calculated to be 8.0×10^{-7} per year, the same as the core damage frequency increase. The percent increase in the large early release frequency is 106.0%.

From Table 1 (main report), Categorization of Risk Impacts Affecting Core Damage Frequency, a 21.1% increase in the core damage frequency corresponds to a category of Medium. However, the categorization

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[2] Successful isolation of a RWCU pipe break is assumed to result in an isolation transient which is not modeled due to the low probability of occurrence compared with other isolation transients. Following successful isolation, no other equipment failures are expected due to initiator.

of the increase in the large early release frequency, from Table 2 (main report) results in a risk impact category of High based on a 106% increase in LERF.

Sensitivity Studies

It should be noted, that the results above are dominated by the operator action to isolate the RWCU system. It is therefore prudent to perform sensitivity cases on the issue with attention to the dominant contributor. The operator action failure rate used in the initial case is $1x10^{-2}$. This value is conservative given the changes made to alarm response procedures and emphasis on operator response training. The sensitivity case (case 2) uses an operator failure rate of $1x10^{-3}$ to estimate the frequency of RWCU Unisolated Pipe Breaks and impacts on core damage and large early release frequency. The estimation of core damage frequency and large early release frequency is performed as above and presented as Case 2 on Table B-11 and Table B-12.

Case 2 results in a core damage frequency increase of 3.2% and a large early release frequency increase of 16.3%. Base on core damage frequency the risk impact category is low and based on large early release frequency increase the risk impact category is medium.

Results and Conclusions

The risk impact of the deferral of the RWCU LOCA Detection and Isolation Modification is presented in Table 13, below. The results indicate that although the core damage frequency increase remains relatively low the large early release frequency could experience a significant increase. Using conservative values in the evaluation of the risk impact of the RWCU unisolated break, the increase in large early release frequency is 106%. With less conservative values the large early release frequency increase is approximately 16%. The risk impact is dominated by operator action error rates.

Based on the degradation of the a significant fission product barrier, and the significant increase in the large early release fraction it is recommended that the RWCU system modification to install LOCA detection and monitoring proceed as originally scheduled.

Table B.13 –
RWCU Unisolated Pipe Break
Core Damage and Large Early Release Frequency Results

Case Description	Core Damage Frequency		Large Early Release Frequency	
	Value	Percent Increase	Value	Percent Increase
Case 1: Unisolated Pipe Break, Operator Action Equal to 0.01	$8.0x10^{-7}$	21.1%	$8.0x10^{-7}$	106%
Case 2: Unisolated Pipe Break, Operator Action Equal to 0.001	$1.2x10^{-7}$	3.2%	$1.2x10^{-7}$	16%

TABLE B-10:
ESTIMATION OF THE PROBABILITY OF
OF AN UNISOLATED RWCU LINE BREAK

RWCU LOCA FREQUENCY ESTIMATION				
No.	Event Description	Reference	Case 1	Case 2
A.1	Generic Pipe Break Frequency (per section per hours)	1	8.60E-10	8.60E-10
A.2	Number of Hours in a Year	n/a	8.76E+03	8.76E+03
A.3	Annual Pipe Break Frequency (per pipe section)	= A.1 * A.2	7.53E-06	7.53E-06
A.4	Estimated No. of Pipe Sections	2	10	10
A.5	**Total Pipe Break Frequency**	**= A.2 * A.4**	**7.53E-05**	**7.53E-05**

ISOLATION PROBABILITY ESTIMATION				
No.	Event Description	Reference	Case 1	Case 2
B.1	Operator Isolate RWCU Break	3	1.00E-02	**1.00E-03**
B.2	Single MOV Operates on Demand	1	4.30E-03	4.30E-03
B.3	Beta Factor for Two MOVs Fail to Operate on Demand (generic)	4	7.00E-02	7.00E-02
B.4	Two MOVs Fail to Operate on Demand (non-common cause)	= (B.2 - (1 - B.3) ^ 2	1.60E-05	1.60E-05
B.5	Two MOVs Fail to Operate on Demand (common cause)	= B.2 * B.3	3.01E-04	3.01E-04
B.6	RWCU Suction/Discharge MOVs Failure to Isolate on Demand	= B.4 + B.5	3.17E-04	3.17E-04
B.7	Either RWCU Suction or Discharge MOVs Fail to Isolate on Demand	= B.6 * 2	6.34E-04	6.34E-04
B.8	**Failure to Isolate Probability**	**= B.1 + B.7**	**1.06E-02**	**1.63E-03**

C.1	**TOTAL UN-ISOLATED LOCA FREQUENCY**	**= A.5 * B.8**	**8.01E-07**	**1.23E-07**

References:

1. PLG, Incorporated, "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants (Failure Data)", PLG-0500, Volume 2, Revision 0, July 1989.
2. GPU Drawings, BR-2143, BR-2144, BR-M565, and 3E-215-A2-1001.
3. Estimated based on remote (control room) action which proceduralized and trained with approximately 10 minutes for completion.
4. PLG, Incorporated, "Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants (Common Cause Failure)", PLG-0500, Volume 4, Revision 1, July 1989.

Table B-11: RWCU LERF Estimation Spreadsheet

Reference Case:	Base Case (Risk Model: OCPRA-13)
Case under study:	RWCU Line Break (CASE 1)

Level 1 - Initiating Events

I.E.	Value	Reference	Variance
CMSIV	4.17E-01	4.17E-01	0%
EPRH	5.61E-02	5.61E-02	0%
EPRL	1.76E-01	1.76E-01	0%
IADS	1.33E-03	1.33E-03	0%
IEMRV	3.31E-02	3.31E-02	0%
LAICS	8.21E-05	8.21E-05	0%
LAIMS	1.15E-04	1.15E-04	0%
LAOIC	6.96E-08	6.96E-08	0%
LAOMS	6.44E-08	6.44E-08	0%
LBI	5.67E-04	5.67E-04	0%
LBIO	8.37E-06	8.37E-06	0%
LOCV	2.24E-01	2.24E-01	0%
LOCW	2.71E-02	2.71E-02	0%
LOFC	1.71E-01	1.71E-01	0%
LOFW	1.51E-01	1.51E-01	0%
LOIA	4.33E-02	4.33E-02	0%
LOIS	7.51E-03	7.51E-03	0%
LOSP	3.26E-02	3.26E-02	0%
LOTB	1.03E-02	1.03E-02	0%
PLOFW	1.78E-01	1.78E-01	0%
RT	7.21E-01	7.21E-01	0%
SAI	9.27E-03	9.27E-03	0%
SAOIC	1.59E-06	1.59E-06	0%
SAORB	7.70E-07	7.70E-07	0%
SAOTB	3.64E-04	3.64E-04	0%
SBI	7.81E-03	7.81E-03	0%
SBO	2.86E-06	2.86E-06	0%
TTRIP	8.97E-01	8.97E-01	0%

Level 1 - Key Plant Damage States

Input KPDS	Input Value	Reference *Base Case*	Percent Variance
PIFW	1.16E-06	1.16E-06	0%
NIFW	1.04E-06	1.04E-06	0%
OIAU	5.75E-07	5.75E-07	0%
OJAU	9.84E-07	1.83E-07	438%
MKCU	1.72E-07	1.72E-07	0%
MJAU	5.88E-08	5.88E-08	0%
NJHW	1.56E-08	1.56E-08	0%
Total CDF	4.60E-06	3.80E-06	0%

LERF Estimation

	Percent of CDF Analyzed =		87.08%
	Total analyzed frequency =		4.00E-06
Category 1A - Large Early			
MKCU	100%	1.72E-07	1.72E-07
NIFW	30.85%	1.04E-06	3.22E-07
OIAU	0.95%	5.75E-07	5.47E-09
		Total	4.99E-07
	Percent of Total Analyzed =		12.46%
Category 1B - Containment Bypass			
OJAU	100%	9.84E-07	9.84E-07
MJAU	100%	5.88E-08	5.88E-08
NJHW	100%	1.56E-08	1.56E-08
		Total	1.06E-06
	Percent of Total Analyzed =		26.43%
Total LERF (sum of above) =			1.56E-06
		Reference LERF =	7.56E-07
		Percent Change in LERF =	**105.93%**
		Percent of Total Analyzed =	38.88%

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Risk significant cutoffs:	Risk Significant Cutoff	Delta for this Case
CDF	51.40%	21.09%
LERF	36.37%	105.93%

Comments: **High Risk Significance**

Table B-12: RWCU LERF Estimation Spreadsheet

Reference Case:	Base Case (Risk Model: OCPRA-13)
Case under study:	RWCU Line Break (CASE 2)

Level 1 - Initiating Events

I.E.	Value	Reference	Variance
CMSIV	4.17E-01	4.17E-01	0%
EPRH	5.61E-02	5.61E-02	0%
EPRL	1.76E-01	1.76E-01	0%
IADS	1.33E-03	1.33E-03	0%
IEMRV	3.31E-02	3.31E-02	0%
LAICS	8.21E-05	8.21E-05	0%
LAIMS	1.15E-04	1.15E-04	0%
LAOIC	6.96E-08	6.96E-08	0%
LAOMS	6.44E-08	6.44E-08	0%
LBI	5.67E-04	5.67E-04	0%
LBIO	8.37E-06	8.37E-06	0%
LOCV	2.24E-01	2.24E-01	0%
LOCW	2.71E-02	2.71E-02	0%
LOFC	1.71E-01	1.71E-01	0%
LOFW	1.51E-01	1.51E-01	0%
LOIA	4.33E-02	4.33E-02	0%
LOIS	7.51E-03	7.51E-03	0%
LOSP	3.26E-02	3.26E-02	0%
LOTB	1.03E-02	1.03E-02	0%
PLOFW	1.78E-01	1.78E-01	0%
RT	7.21E-01	7.21E-01	0%
SAI	9.27E-03	9.27E-03	0%
SAOIC	1.59E-06	1.59E-06	0%
SAORB	7.70E-07	7.70E-07	0%
SAOTB	3.64E-04	3.64E-04	0%
SBI	7.81E-03	7.81E-03	0%
SBO	2.86E-06	2.86E-06	0%
TTRIP	8.97E-01	8.97E-01	0%

Level 1 - Key Plant Damage States

Input KPDS	Input Value	*Reference Case*	Percent Variance
PIFW	1.16E-06	1.16E-06	0%
NIFW	1.04E-06	1.04E-06	0%
OIAU	5.75E-07	5.75E-07	0%
OJAU	3.06E-07	1.83E-07	67%
MKCU	1.72E-07	1.72E-07	0%
MJAU	5.88E-08	5.88E-08	0%
NJHW	1.56E-08	1.56E-08	0%
Total CDF	3.92E-06	3.80E-06	3%

LERF Estimation

	Percent of CDF Analyzed =		84.84%
	Total analyzed frequency =		3.33E-06
Category 1A - Large Early			
MKCU	100%	1.72E-07	1.72E-07
NIFW	30.85%	1.04E-06	3.22E-07
OIAU	0.95%	5.75E-07	5.47E-09
		Total	4.99E-07
	Percent of Total Analyzed =		14.99%
Category 1B - Containment Bypass			
OJAU	100%	3.06E-07	3.06E-07
MJAU	100%	5.88E-08	5.88E-08
NJHW	100%	1.56E-08	1.56E-08
		Total	3.80E-07
	Percent of Total Analyzed =		11.43%
Total LERF (sum of above) =			8.79E-07
	Reference LERF =		7.56E-07
	Percent Change in LERF =		**16.27%**
	Percent of Total Analyzed =		26.43%

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Risk significant cutoffs:	Risk Significant Cutoff	Delta for this Case
CDF	51.40%	3.24%
LERF	36.37%	16.27%

Comments: **Less Conservative Human Action Value**
Medium Risk Significance

APPENDIX C

FIRE INDIVIDUAL PLANT EXAMINATION

METHODOLOGY OVERVIEW

4.0 Oyster Creek Fire Individual Plant Examination

The Oyster Creek Fire Individual Plant Examination report presents the methods and results of the fire analysis of the impacts of fire events at the Oyster Creek Nuclear Generating Station (OCNGS).

The study is performed in response to the Nuclear Regulatory Commission's (NRC) Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities". The analysis satisfies the requirement for the Internal Fire Analysis and presents the methods, calculations and results in the suggested NUREG-1407 format.

The analysis is performed using standard probabilistic methods similar to those used in the development of a Level 1 Probabilistic Risk Assessment with several notable exceptions.

- First, all accident sequences developed in this study are initiated by fire events which are internal to the plant.
- Second, a cutoff in the frequency of core damage is used to screen fire areas and hence the study is termed the Oyster Creek Individual Plant Examination or a scoping studying. One of the outcomes of using a screening approach is that the core damage frequency reported represents an upper bound since a more detailed evaluation would result in lower core damage contributions of individual fire areas. This approach to analyzing internally initiated fire events is a less resource intensive effort while still providing assurance that plant specific vulnerabilities, if any, are determined.
- Third, significant portions of the Electric Power Research Institutes (EPRI) Fire Induced Vulnerability Evaluation (FIVE) methods are used in the study.

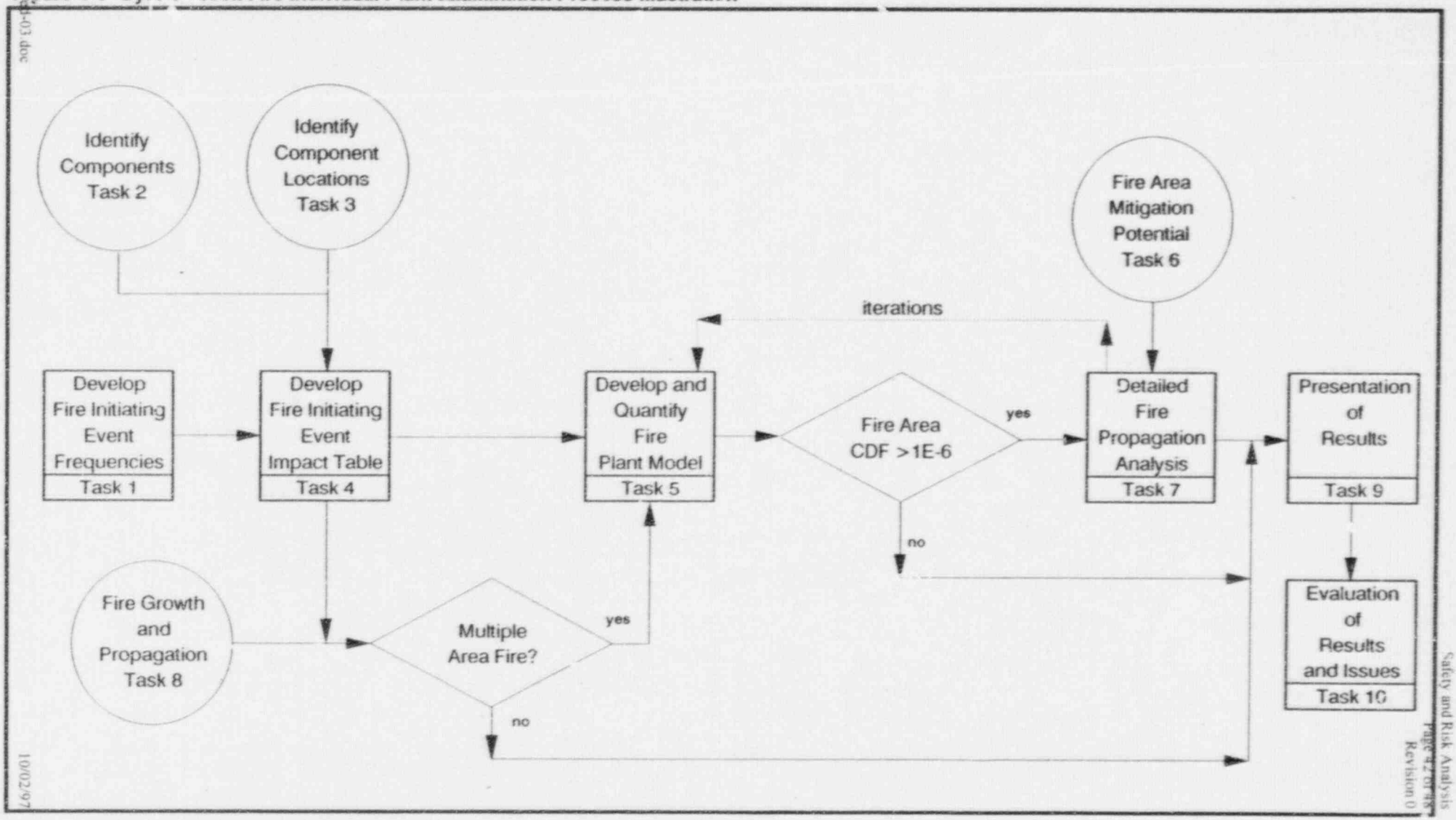
The study is comprised of ten tasks and results in the evaluation of the risk of internal fires at the Oyster Creek Nuclear Generating Station (OCNGS). The process is described in overview in the following paragraphs and illustrated in Figure 4-1. It should be noted that since the report is organized in the suggested NUREG-1407 format, multiple tasks are often documented in a single report section or sub-section. Each task or group of tasks as illustrated on Figure 4-1 is described as well as illustrated in the associated figure to the right.

Task 1 - Develop Fire Initiating Event Frequencies

This task identifies areas of the Oyster Creek Nuclear Generating Station in which the potential for fire initiation, growth and/or propagation can significantly impact plant operation from at power conditions.

The input to this task is from the Level 1 Oyster Creek Probabilistic Risk Assessment (OCPRA), the Fire Hazard Analysis Report, Oyster Creek Fire Mitigation Procedure and plant walkdowns.

Figure 4-1 Oyster Creek Fire Individual Plant Examination Process Illustration



In this task the Fire Hazard Analysis Report fire area and zone designations are used with the OCPRA, fire mitigation procedure and plant walkdowns to determine plant areas in which a fire event may perturb plant operation sufficiently to result in a demand for a reactor scram. The Electric Power Research Institutes (EPRI) Fire Induced Vulnerability Evaluation (FIVE) methodology and database is then used to develop fire initiating event frequencies for each of the identified fire areas and zones (critical fire areas).

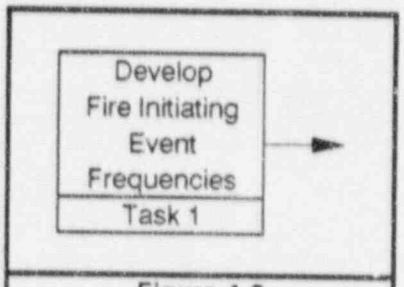


Figure 4-2
Develop Fire Initiating Events

Several fire areas are screened from further consideration based on the insignificant impact of the fire event. For example, the Site Emergency Building does not contain any plant equipment and is located some distance from the plant and, as such, a fire in this area is not expected to result in a demand for a plant trip or damage to plant equipment.

The list of fire areas and zones with their frequency of fire ignition serves as input to the development of Task 4 (Development of the Fire Initiating Event Impact Table) and Task 5 (Development and Quantification of the Plant Model). Details on this task are presented in report Section 4.1, Fire Hazard Analysis.

Task 2 - Identification of Risk Significant Components

In this task the Level 1 OCPRA, the Fire Hazard Analysis Report (FHAR) and plant walkdowns as well as the list of critical fire areas (from Task 1) are used to determine the potential risk significant components. These components are then screened on the basis of their susceptibility to fire events. The result is the list of risk significant components. Details on this task are presented in report Section 4.4, Evaluation of Component Fragilities and Failure Modes.

Task 3 - Identification of Risk Significant Component Locations

In this task, reviews of plant information and plant walkdowns are used to determine the location of the risk significant components and their supporting cables.

Supporting cables include any required electrical or other functional system support cables. Supporting cables also include the possibility of component failure due to "hot shorts" which cause the component to go to an active failure position. That is, supporting cables include those cables whose electrical hot short (i.e., energized) can result in a component changing state into an undesired state or position. For example, a normally closed valve changing to the open position due to a fire event which affects the cable in a remote location of the plant.

The result of this task is the Location of Risk Significant Components and Associated Cables Table, which is used in the Development of the Fire Initiating Event Impact Table (Task 4).

Several fire areas are screened from further consideration since they contain no risk significant components or supporting cables. These areas are screened from further consideration for their individual contribution to the core damage frequency however they are still considered for their

potential to be involved in multiple area fires (Task 8).

The information developed in this task is used as input for the development of the Fire Initiating Event Impact Table (Task 4) and the Fire Growth and Propagation (Task 8). Details on Task 3 are presented in report Section 4.4, Evaluation of Component Fragilities and Failure Modes.

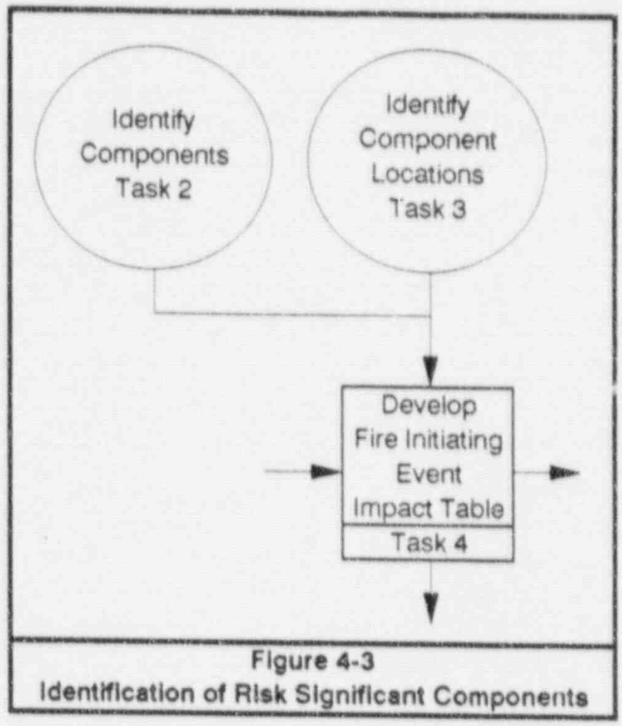


Figure 4-3
Identification of Risk Significant Components

Task 4 - Develop Fire Initiating Event Impact Table

In this task the Location of the Risk Significant Components and Associated Cables (Task 3) and the OCPRA are used to develop the Fire Initiating Event Impact Table.

Each critical fire area, as defined in Task 1, is considered an initiating event. Using the physical component locations and the locations of supporting cables a five event impact table can be developed. This impact table provides the affected components (an hence system functions) given an "all engulfing fire" within a fire area. The term "all engulfing fire" is used to describe the modeling of a fire which fails all components and cables in the area and does not account for detection, suppression or other area mitigative features. In addition, "hot short" impacts are included in the impact table. The Fire Initiating Event Impact Table is therefore the most conservative impacts which a fire event within a given fire area can cause.

The impact table is used as input into the Development and Quantification of the Fire Risk Model (Task 5). Details on Task 4 are presented in report Section 4.2, Review of Plant Information and Walkdowns.

Task 5 - Development and Quantification of the Plant Model

This task develops and documents the Oyster Creek Fire Risk Model. Actually three sub-tasks are performed in the development and quantification of the fire risk model and these sub-tasks are represented on Figure 4-4 as three separate paths of input and output. All three sub-tasks are documented in report Section 4.6, Analysis of Plant Systems, Sequences and Plant Response.

- The first input/output path develops the individual fire area upper bound core damage frequency estimations with input from Tasks 1 and 4 and is described in the "*Initial Estimate of Upper Bound Core Damage Frequency*" report sub-section.

- The second input/output path is represented as the iteration loop between the Detailed Fire Propagation Analysis (Task 7) and develops the refined core damage frequency estimates for those fire areas whose upper bound core damage frequency (UBCDF) was initially greater than $1x10^{-6}$. A single iterations is made which results in the calculation of the Revised Estimate of Upper Bound Core Damage Frequency. Any fire areas which are not screened (UBCDF less than $1x10^{-6}$) are analyzed in Task 7 and documented in the "*Detailed Evaluation of Core Damage Frequency*" report sub-section.

- The third input/output path develops the "multiple fire area" upper bound core damage frequency estimations.

Each input/output paths is discussed in detail below.

Initial Estimate of Upper Bound Core Damage Frequency (UBCDF)

The first input/output path used the Level 1 OCPRA, Fire Initiating Event Frequencies (Task 1) and the Initiating Event Impact Table (Task 4) to develop and quantify the fire risk model for the "*Initial Estimation of the Upper Bound Core Damage Frequency*" as a result of fire events within an individual fire area.

The impacts of a fire event (Task 4) together with the fire initiating event frequency (Task 1) are combined with the random failure probabilities of system functions modeled in the Level 1 OCPRA to produce the fire risk model. That is, the failures produced by the fire initiating event are added to the OCPRA plant model (the independent failures) to produce a risk model which calculates the core damage frequency due to fire events.

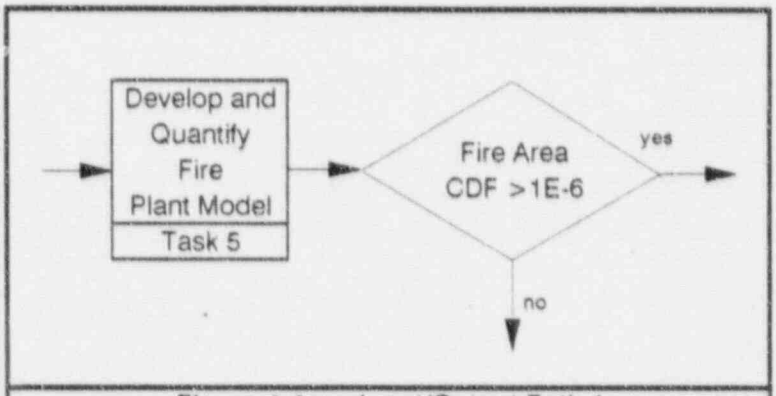


Figure 4-4a - Input/Output Path 1
Initial Estimation of Upper Bound Core Damage Frequency

Since the fire initiating event impact table represents the most conservative outcome of a fire in a given fire area (i.e., "all engulfing fire" and "hot shorts") and fire growth, propagation, detection, suppression or other fire area mitigative features are not modeled, the quantification of this fire risk model produces an upper bound core damage frequency for each fire event. Fire areas whose UBCDF is less than $1x10^{-6}$ per year are screened from further consideration. Fire areas whose total UBCDF contribution is greater than $1x10^{-6}$ per year require a Revised Estimate of Upper Bound Core Damage Frequency which is performed as part of the input/output path two, described below.

Revised Estimate of Upper Bound Core Damage Frequency

In the second input/output path the fire areas whose initial upper bound core damage frequency was greater that $1x10^{-6}$ per year are evaluated. Assumptions regarding the "all engulfing fire" and fire risk model simplifications are addressed and potentially relaxed to more accurately reflect the risk associated with a fire event in these particular fire areas.

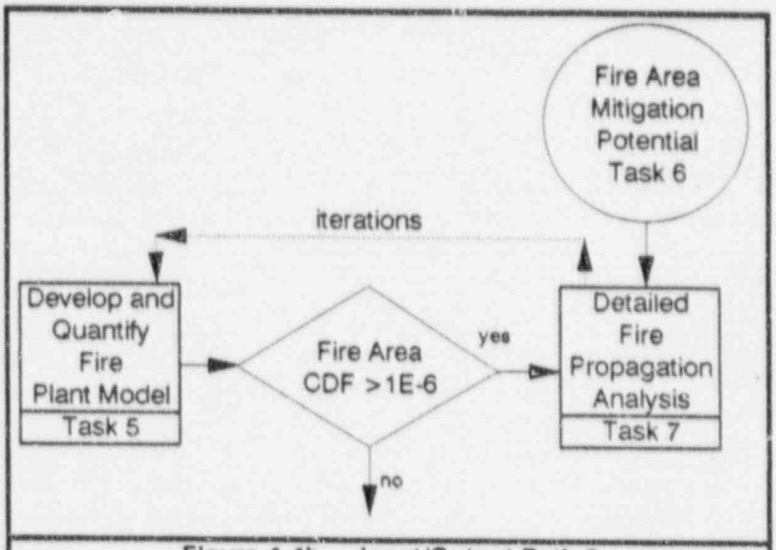


Figure 4-4b - Input/Output Path 2
Revised Estimation of Upper Bound Core Damage Frequency

Following the adjustment of the conservative assumptions the fire risk model is requantified. In the case where the total fire area UBCDF is less than $1x10^{-6}$ per year the fire area is screened from further consideration.

Where the total fire area UBCDF is greater than $1x10^{-6}$ per year the output is directed to Task 7, Detailed Evaluation of Fire Core Damage Frequency. This sub-task is documented in report Section 4.6.2, Revised Estimation of Upper Bound Core Damage Frequency.

Upper Bound Core Damage Frequency Estimation for Multiple Area Fires

The third input/output path develops and quantifies the fire risk model for multiple fire area events. Input is from the Fire Growth and Propagation Task (Task 8), the Development of the Fire Initiating Event Frequencies (Task 1) and the Fire Initiating Event Impact Table. For each multiple fire area event the frequency of the initiating event is calculated as the sum of the individual fire areas which comprise the event. The impacts of the newly defined initiators are also the sum of the impacts of the individual fire areas which comprise the multiple area fire. The impacts and frequencies are factored into the Level 1 OCPRA. The quantification of this fire risk model produces an estimation of the upper bound core damage frequency as a result of multiple fire area events. This

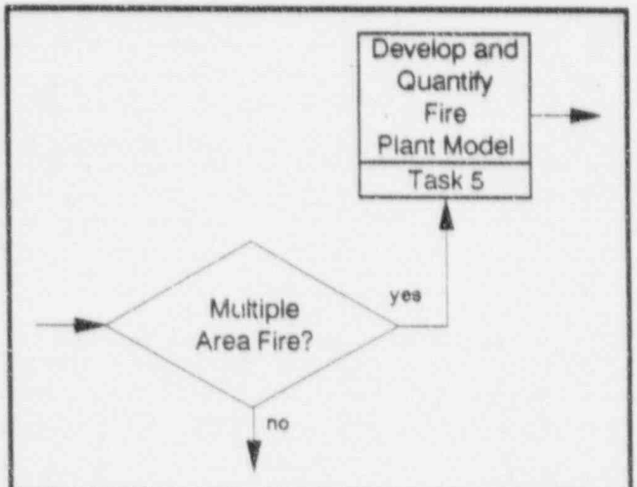


Figure 4-4c - Input/Output Path 3
Multiple Fire Area Quantification

input/output path is documented in report Section 4.3.

Task 6 - Critical Fire Area Mitigation Potential

This task documents the Oyster Creek Nuclear Generating Station's fire detection and suppression systems. Input to the task is from the Fire Hazard Analysis Report and the Fire Mitigation Procedure. The information developed in this task serves as input to the Detailed Fire Propagation Analysis (Task 7). Details on this task are contained in report Section 4.5, Fire Detection and Suppression.

Task 7 - Detailed Fire Propagation Analysis

Those fires areas whose upper bound core damage frequency is greater than $1x10^{-6}$ serve as input into the Detailed Fire Propagation Analysis. The Fire Area Mitigation information collected in Task 6 is used to adjust the conservative assumptions made in the risk model for these areas. The model is then re-quantified. The result of this task is revised risk model impacts and/or adjusted severe fire frequencies. Details on Task 8 are provided in report Section 4.6, Analysis of Plant Systems, Sequences and Plant Response.

Task 8 - Fire Growth and Propagation

This task investigates the potential for fire growth and propagation of fires beyond individual fire areas. Evaluations of fire growth and propagation within a fire area are addressed in the Detailed Fire Propagation Analysis (Task 7) which is presented in report Section 4.6. This task, Fire Growth and Propagation beyond individual fire areas is addressed qualitatively using the Electric Power Research Institutes (EPRI) Fire Induced Vulnerability Evaluation (FIVE) assumptions regarding the effectiveness of fire barriers are applied.

The input to the task is from the Identification of Critical Fire Areas (Task 1) and the Fire Initiating Event Impact Table (Task 4). The result of this task is an evaluation of the potential "multiple area fires". In the case where a multiple area fire is assumed to occur, a new initiating event is developed. This initiating event is equal in frequency of occurrence to the sum of the frequency of fire initiation of the fire areas involved. The impacts of this new initiator is equal to the combined impacts of the fire areas involved. This new initiating event is input into the Development and Quantification of the Fire Risk Model (Task 5). Details on Task 8 are presented in report Section 4.3, Fire Growth and Propagation.

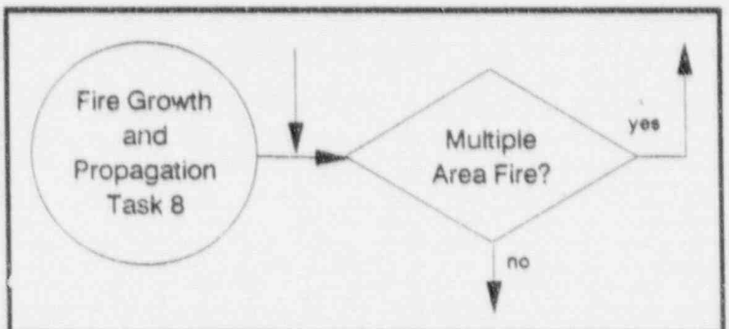


Figure 4-5
Multiple Area Fire Evaluation

Task 9 - Presentation of Results

This task assembles, summarizes and presents the overall results of the Oyster Creek Fire Individual Plant Examination including a summary of containment performance. Details are presented in report Section 4.7, Presentation of Results.

Task 10 - Evaluation of the Results and Fire Issues

This task applies the results and lessons learned to the Sandia issues, A-45 and others. Details are presented in the following report sections:

- Section 4.7, Containment Failure Modes due to Fires
- Section 4.8, Treatment of Sandia Fire Risk Scoping Study Issues
- Section 4.9, USI A-45 and Requirements of NUREG-1407.

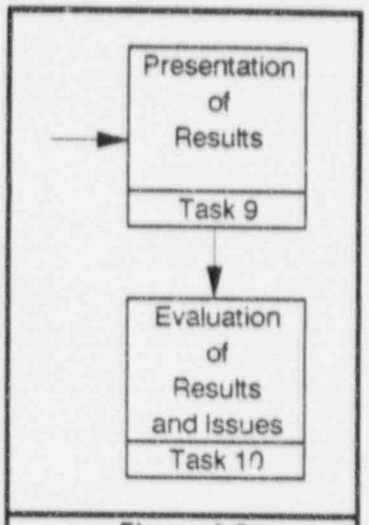


Figure 4-6
Results and Conclusions

Each of the sections of this report begins with a detailed description of the task including the input to the task, output of the task and the steps which are used in the analysis. Taken together, the introduction to each section provides the detailed methodology of the performance of the Oyster Creek Fire Individual Plant Examination.