FINAL REPORT

PHASE II

AN ASSESSMENT OF THE RISK SIGNIFICANCE OF SEP ISSUES FOR OYSTER CREEK

Prepared for

THE NEW JERSEY BUREAU OF RADIATION PROTECTION

By

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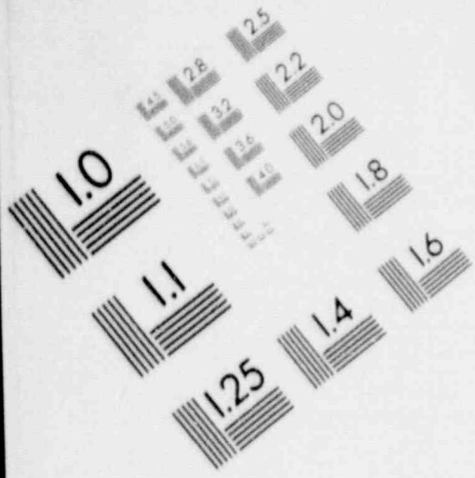

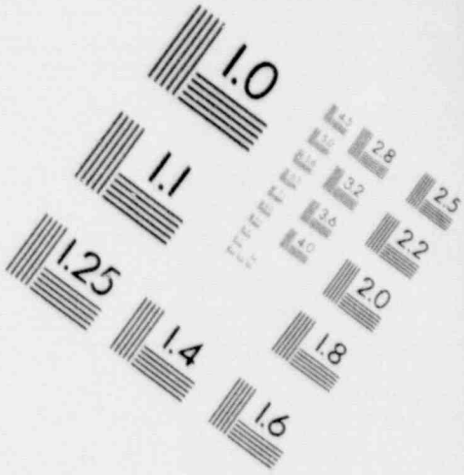

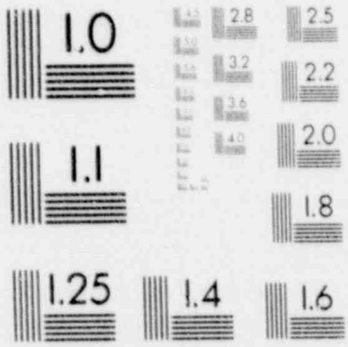

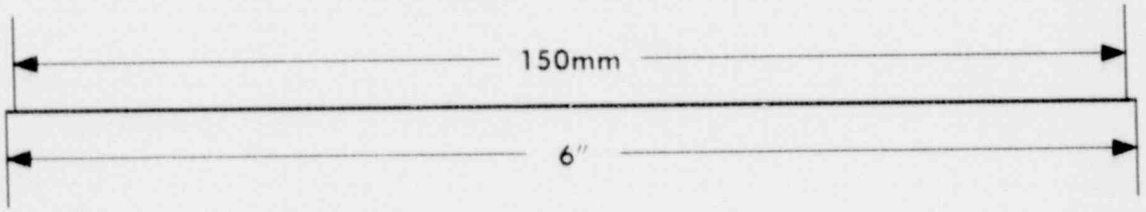

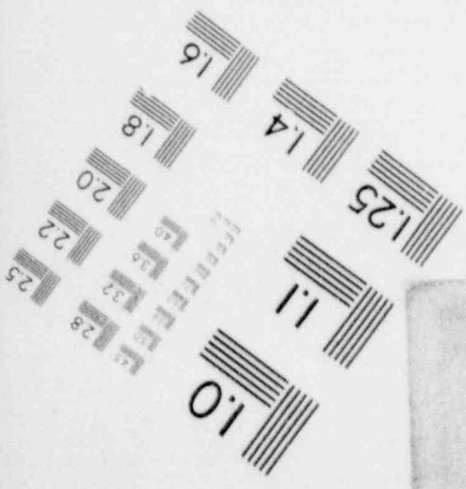

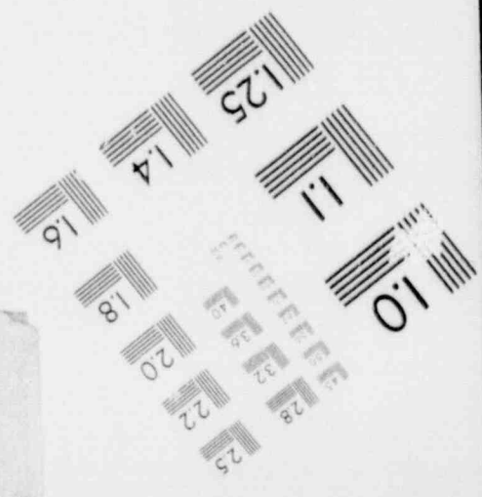

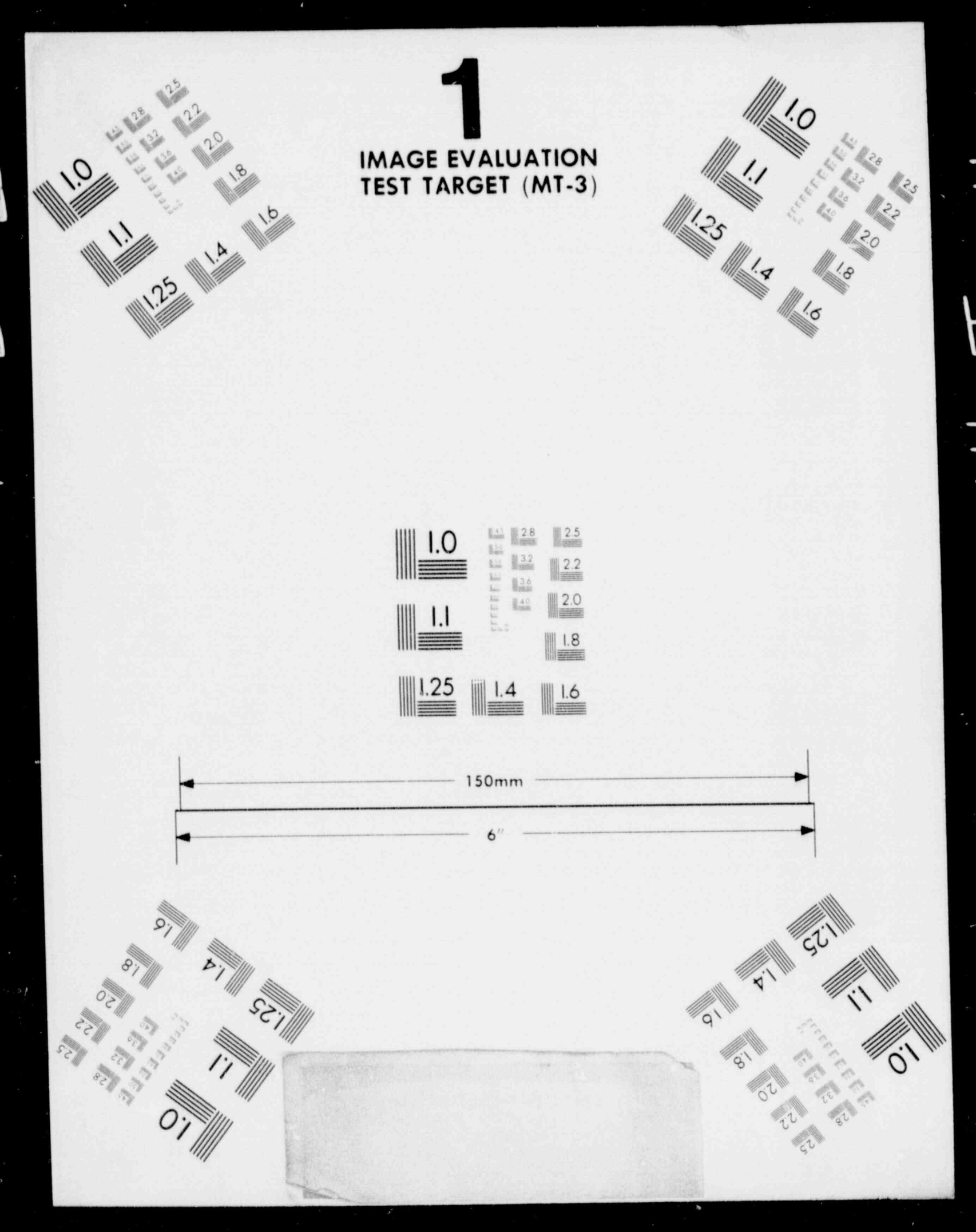
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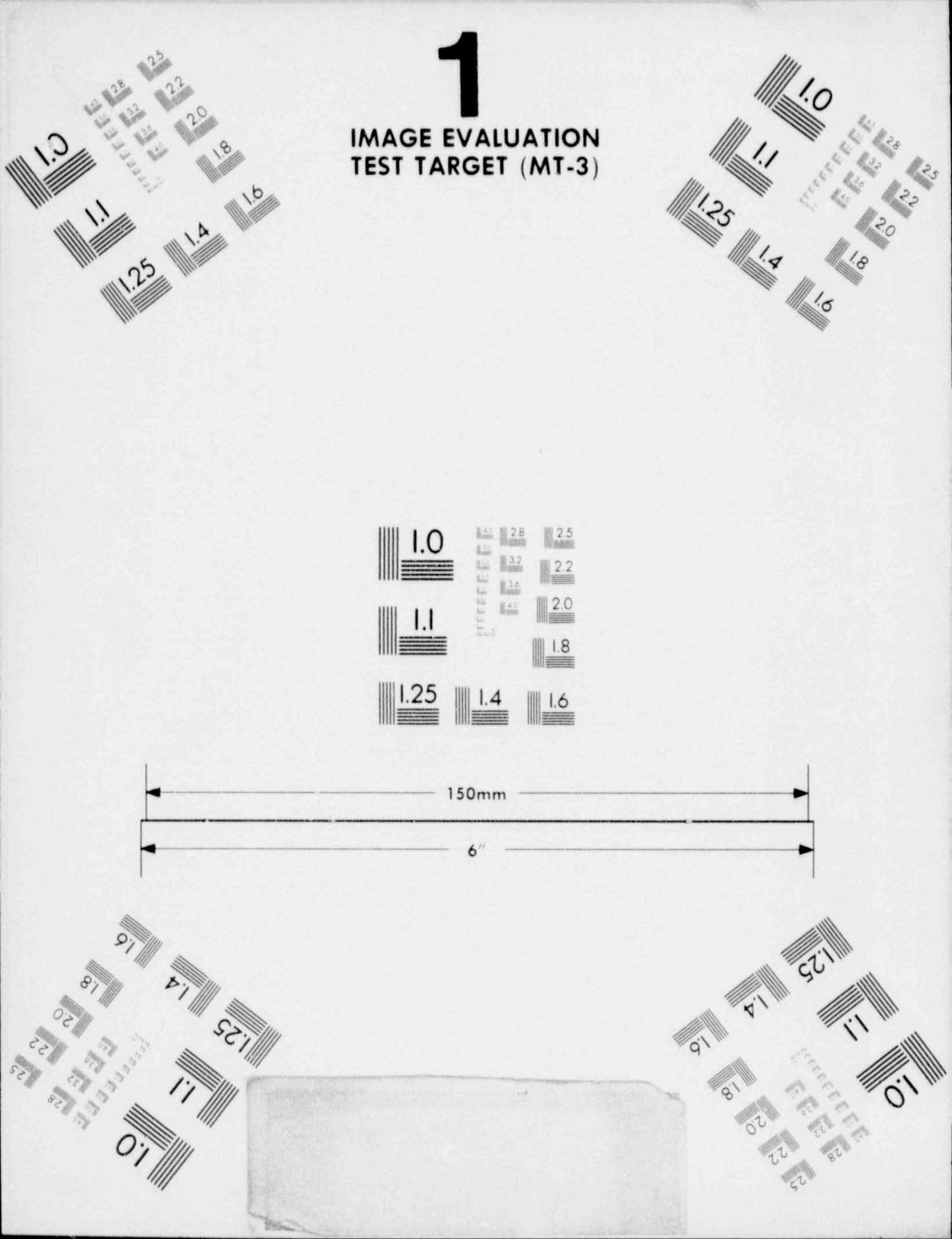
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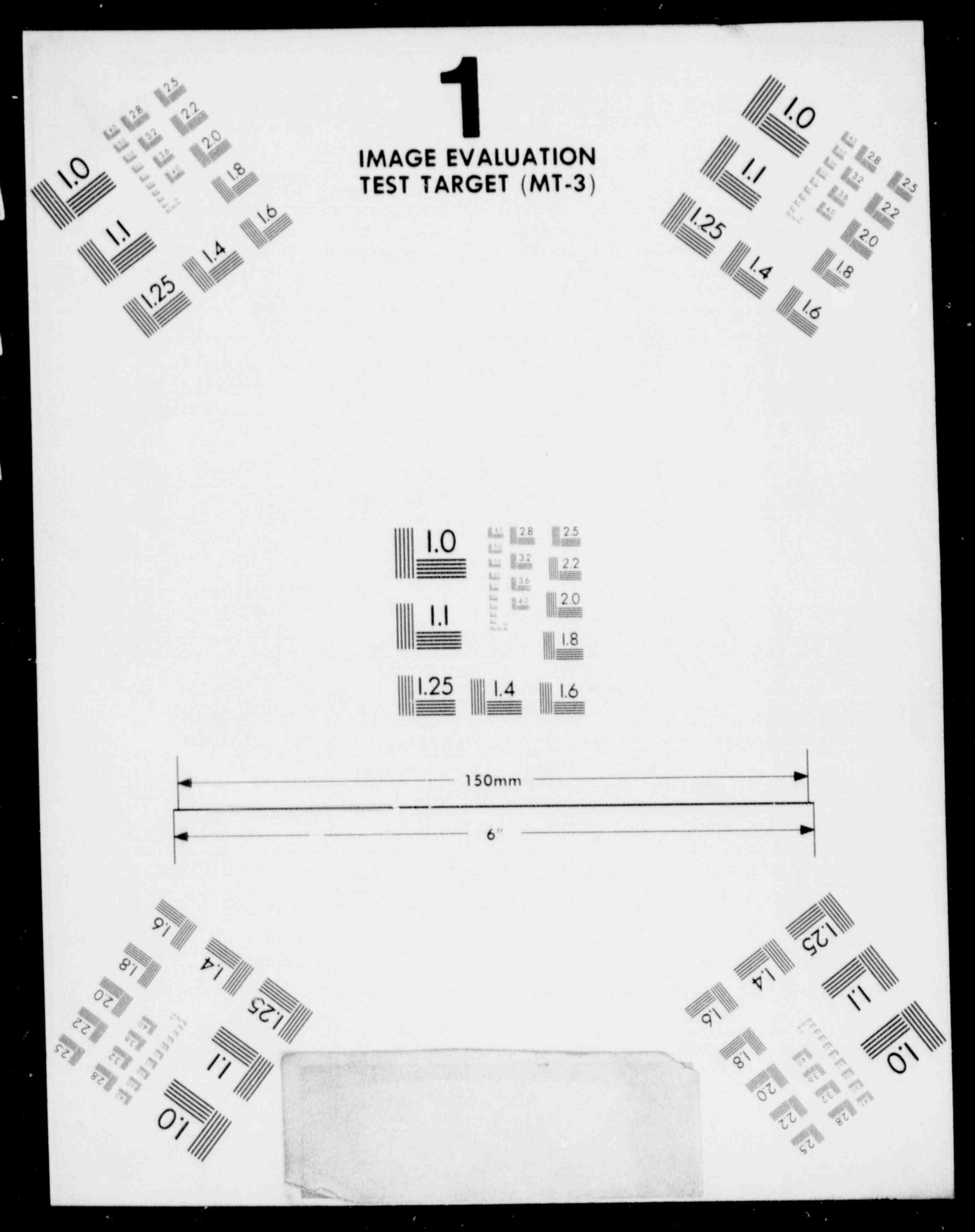
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AN ASSESSMENT OF THE RISK SIGNIFICANCE OF SEP ISSUES FOR OYSTER CREEK

I. INTRODUCTION

This report presents the results of an effort to examine the risk significance of some 87 issues identified by the NRC as areas of potential non-compliance for the Oyster Creek nuclear power plant. The 87 issues were derived by the NRC from some 40 safety-related topics which were found to apply to the Oyster Creek plant during the NRC's Systematic Evaluation Program (SEP). The SEP review involves examining older plants to determine if they are deficient in meeting current NRC regulations which may have been developed since the plant went into operation.

The procedure used in this study to determine the potential risk implication of the 87 issues was as follows:

1. Three Probabilistic Risk Assessment studies for plants similar to Oyster Creek were examined and risk dominant accident sequences identified.

2. A brief survey was undertaken to identify design differences between the three PRA plants and Oyster Creek to determine the extent to which the risk dominant sequences apply.

3. The individual factors comprising the risk dominant sequences were separated to identify risk dominant initiating events and system failures.

4. The major contributors to failure for risk dominant systems were identified.

5. Each of the 87 issues was examined to determine if they would influence the major contributions to failure for the risk dominant systems.

It should be noted that a definitive conclusion regarding the risk significance of the 87 issues could be reached in only a few cases. This is because (1) many of the issues are associated with protection against events postulated to occur external to the plant (tornadoes, earthquakes, floods, etc). None of the PRAs examined for this effort considered external events. Thus, the risk significance of these issues could not be determined within the scope of this effort; (2) the issues were derived from topics which reflect current NRC regulations. These regulations were not developed on the basis of quantitative risk assessment studies, but rather on the conventional NRC approach of deterministic analysis coupled with engineering judgment. Thus, a close coupling between the issues and risk assessment results would not be anticipated and did not, in fact, exist.

II. RISK DOMINANT SEQUENCES FROM PRA STUDIES

There have been six probabilistic risk assessment studies published and made available which consider risks from boiling water reactors. Three of these studies were for reactors with containment designs similar to Oyster Creek (designated Mark I containment). These three were thus selected for use in this study. Of the three, the Millstone Unit 1[1] plant is closest in design details to Oyster Creek, and its use was emphasized in the study. The other two PRA plants, Browns Ferry Unit 1[2] and Peach Bottom Unit 2[3] are of later vintage design than Oyster Creek and have more substantial design differences. The Oyster Creek and Millstone plants commenced commercial operation about one year apart (1969 vs 1970 for Millstone) while Browns Ferry began operation in 1974 as did Peach Bottom. A further discussion of the differences among the plants is provided in Section III.

A Probabilistic Risk Assessment study for a nuclear power plant attempts to quantify public risk from the operation of the plant. This is done by calculating both the probability of serious accidents and their consequences. The product of these two quantities is the accepted definition of risk. Accident

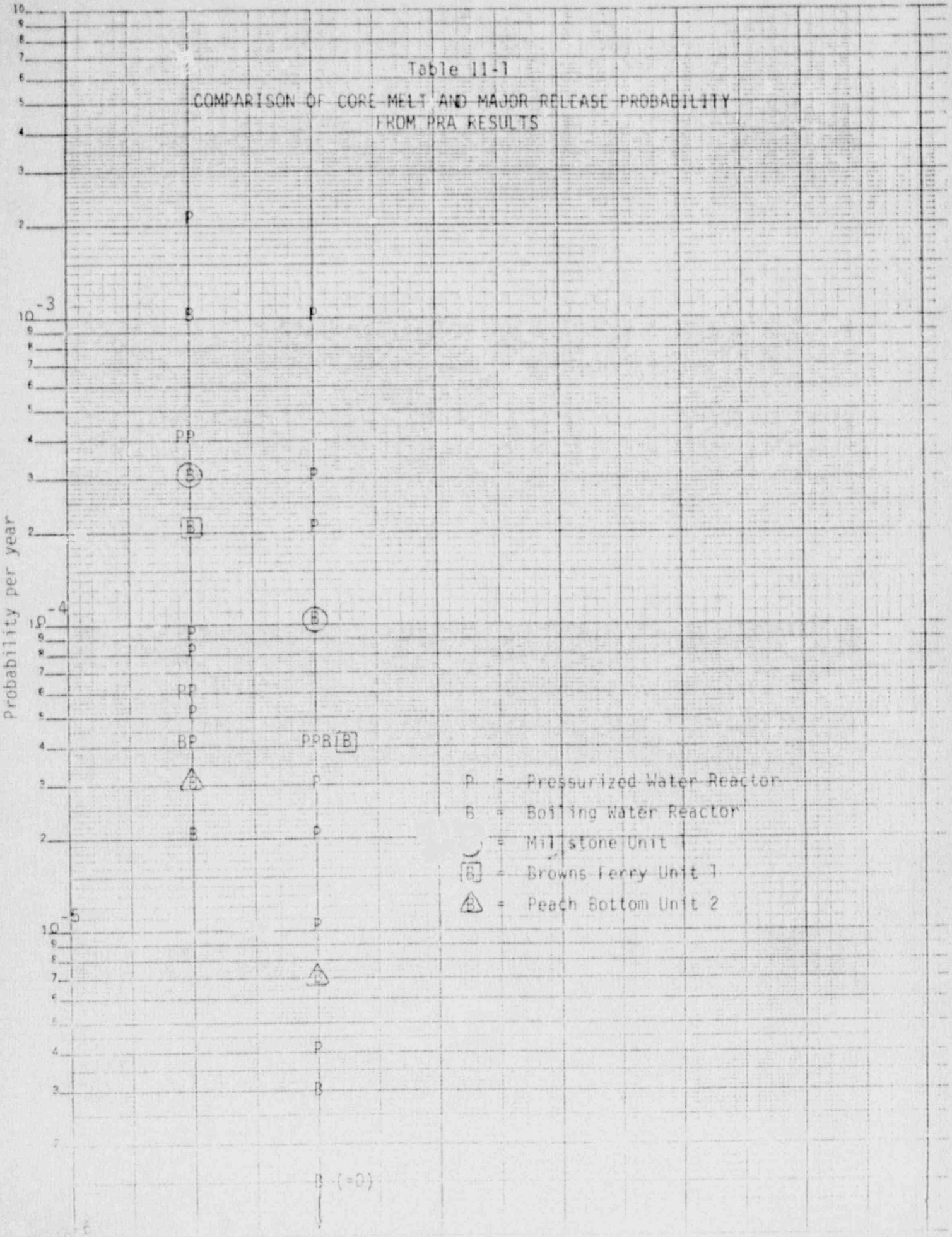
probabilities are calculated by assessing the frequency of accident initiating events and then calculating the probability of failure of those systems which are designed to successfully terminate the consequences of the event. A complete discussion of PRA methodology can be found in Reference 4.

Before discussing the risk dominant accident sequences from the three PRA studies examined for this effort, useful perspective can be obtained by comparing available PRA results. Table II-1 provides this comparison. The vertical scale (logarithmic) shows probability for core melt and major release as listed at the bottom of the table. Core melt probability is the annual probability that an accident resulting in fuel melting was calculated to occur. The probability of major release is the annual probability that the containment will fail catastrophically following a core melt. The symbols "B" and "P" are used to indicate these probabilities, respectively, for boiling and pressurized water reactors as calculated by the 15 available PRA studies[5]. Symbols around the "B" indicate which results are from which of the three PRA studies utilized in this study. As shown in the table, the Millstone Point Unit 1 core melt probability result for a plant which most closely represents Oyster Creek is towards the upper part of the range of results. The frequency of a major release for Millstone is the highest of all BWRs, yet significantly lower (a factor of 10) than the highest PWR results. While the Table II-1 comparisons provide some perspective, the results should be viewed with considerable skepticism for the following reasons: (1) the results were produced at different times (as much as 9 years apart) by different organizations employing different methodologies, and (2) large uncertainties are associated with all results, which tend to diminish the significance of differences.

Returning to the identification of dominant risk sequences from the PRA studies, results from each study will be considered separately. It is important to recognize that none of the three studies considered external events as accident initiators and only the Peach Bottom PRA actually calculated health consequences from the accidents. Rather than specific health consequences, the Millstone and Browns Ferry studies assigned accident sequences to four release categories which were originally defined for Peach Bottom. Each release category has a specific set of characteristics (amount, timing, and elevation) which describe the radionuclide release. The accident sequences are assigned to release

Table 11-1

COMPARISON OF CORE MELT AND MAJOR RELEASE PROBABILITY FROM PRA RESULTS



categories and probabilities computed on the basis of the accident sequence characteristics which are important to the release characteristics. For example, in some accident sequences the containment is predicted to fail before the core melts. For these cases, the accidents are assigned to a high release category since hold-up and retention of fission products in the pre-failed containment would not be as significant as other sequences in which the containment is intact at the onset of core melt.

For each of the three PRA studies, four release categories were found to be adequate to span the range of radionuclide release characteristics for the accident sequences found to be important. In the PRA which considered Peach Bottom[3], a composite reactor site was defined for the purposes of calculating accident consequences. Based on a more explicit consideration of these consequences[6], the Table II-2 results were obtained. This table illustrates the different average consequences in terms of early and latent fatalities computed for each release category for the "composite" site. The table illustrates the rather significant differences in calculated consequences for each release category. For Table II-1, a "major release" was defined as a Category 1, 2, or 3 release. However, since the containment was predicted to fail for accidents in all four categories, a significant release would occur even in Category 4 (and fatalities are predicted to occur as shown in Table II-2), the least extensive release. For these reasons, this study will focus on core melt probability rather than risk, but it should be recognized that some core melt accidents impose more significant public consequences than others. A separate assessment of risks from Oyster Creek[16] has been performed which considers the risk from different types of core melt accidents.

A. Millstone Point Unit 1 Dominant Accident Sequences

Table II-3 shows the frequency of core melt accidents for each of the four release categories discussed previously. The total core melt probability is dominated by accidents predicted to produce Category 3 and 4 releases.

Table II-4 provides a list of the accident sequences which dominate the core melt probability for Millstone. The listed sequences contribute 83% of the total. No other single accident sequence contributes more than 1% to the total.

Table II-2

COMPARISON OF CONSEQUENCES ASSOCIATED WITH RELEASE CATEGORIES USED IN THE BWR PRAs CONSIDERED IN THIS STUDY

Release Category	Average Consequences	
	Early Fatalities	Latent Fatalities
1	4.3	870
2	0.05	533
3	0	290
4	0	39

Table II-3

FREQUENCY OF CORE MELT ACCIDENTS FOR EACH RELEASE CATEGORY FROM THE MILLSTONE POINT UNIT 1 PRA

Category	Frequency, per year
1	1E-6
2	8E-6
3	1E-4
4	2E-4
Total	3E-4

Table II-4

ACCIDENT SEQUENCES WHICH DOMINATE CORE MELT
PROBABILITY FROM MILLSTONE PRA

Sequence	Probability	% Contribution to Total CMP
1. T_4JCD	6.7E-5	22
2. T_4JCEFG	4E-5	13
3. T_4KCEFG	3E-5	10
4. T_4KCD	3E-5	10
5. T_4LCD	3E-5	10
6. T_2A	2E-5	7
7. T_4JCMG	9E-6	3
8. T_4LCMG	9E-6	3
9. T_4KCMG	8E-6	3
10. T_5MG	6E-6	2
Total	3E-4	83

Definition of Symbols:

T_2 - Transient with loss of power conversion system
T_4 - Transient with loss of normal ac power
T_5 - Transient with stuck open relief valve
J - Safety relief valves fail to close
K - Isolation condenser system failure
L - Isolation condenser makeup system failure
A - Reactor protection systems failure
M - Shutdown cooling system failure
C - Feedwater Cooling Injection System failure
G - Containment cooling system failure
E - Low pressure coolant injection system failure
F - Core spray system failure
D - Automatic depressurization system failure

The sequences are designed (first column) by a series of letters which represent an initiating event (first letter) followed by a set of system failures (subsequent letters) which are predicted to result in core melt. The probability of the sequence is the product of the initiating event frequency and the subsequent system failures. The letters are defined at the bottom of the table.

The table illustrates that the most risk significant initiating event is a transient with loss of normal ac power (T_4). The frequency of this transient was assessed (Table 4.1 of Ref. 1) to be 0.20/yr. This is the same value used for Peach Bottom and Browns Ferry and represents a normal ac power loss once every 5 years. To check the validity of this frequency for Oyster Creek, Appendix F of the SEP[7] was examined. Appendix F contains a review of Oyster Creek operating data and includes events which have occurred at the plant. Although the Appendix F results are somewhat confusing, it appears (Table 4.2) that two normal ac power disturbances resulting in forced shutdown have occurred at Oyster Creek in the time period 1969-1981. This is consistent with a 0.2/yr frequency of normal ac power loss.

In order to assess the significance of the 87 SEP issues on core melt probability, it is necessary to examine not only the risk significant systems but also the major causes of system failure. An SEP issue may influence the design or operation of a risk dominant system, but if it doesn't affect the major causes of system failure in the risk dominant sequences, it will not have an influence on risk reduction.

From the Millstone PRA, Table II-5 was compiled which indicates the dominant contributors to failure for each of the risk dominant systems identified in Table II-4.

B. Browns Ferry Unit 1 Dominant Accident Sequences

Table II-6 lists the three dominant accident sequences from the Browns Ferry PRA study which collectively contribute about 90% to the total core melt probability of 2E-4. All other sequences found in the study individually contribute

Table I.-5

MAJOR CAUSES OF FAILURE FOR RISK DOMINANT SYSTEMS
FROM MILLSTONE PRA RESULTS

System	Major Failure Causes	Comments
Safety relief valves (J)	direct from data	no causes identified
Isolation Condenser (K)	isolation valves closed for maintenance	valve failures contribute about 70%
Isolation Condenser Makeup (L)	failure of makeup valve	
Reactor Protection (A)	mechanical failure of rods	common mode failure
Shutdown Cooling (M)	breaker/transformers auto bus transfer	see (G) below
Feedwater Coolant Injection (C)	gas turbine breaker pump breaker gas turbine generator service water system condensate transfer system relays pump permissive switches	listed in decreasing order of significance
Containment Cooling (G)	breaker/transformer auto bus transfer	same failures as for (M) above
Low Pressure Coolant Injection (E)	gas turbine and diesel generator gas turbine breaker and diesel generator trip reset logic for emergency power	
Core Spray Injection (F)	(same as above)	
Automatic Depressurization (D)	human error	for the risk significant sequences, this system will not automatically actuate because drywell pressure does not increase. Thus, manual actuation is necessary.

Table II-6

ACCIDENT SEQUENCES WHICH DOMINATE CORE MELT
PROBABILITY FROM BROWNS FERRY PRA

Sequence	Probability	% Contribution to Total
1. $T_u R_B R_A$	9.7E-5	49
2. $T_u B$	5.1E-5	26
3. $T_P R_B R_A$	2.8E-5	14
Total	1.8E-4	89

Definition of Symbols:

T_u - Transient with power conversion system unavailable

T_p - Loss of offsite power transient

R_B - Residual heat removal (torus cooling)

B - Control rod drive system

R_A - Residual heat removal (shutdown cooling)

Table II-7

MAJOR CAUSES OF FAILURE FOR RISK DOMINANT
SYSTEMS FROM BROWNS FERRY PRA RESULTS

Sequence	System	Major Failure Causes	Comments
$T_u R_B R_A$	RHR-torus cooling (R_B)	Valve control circuit faults	Both R_B and R_A are failed by the valve control circuit fault
$T_u B$	Control rod drive (B)	scram failure	
$T_u R_B R_A$	RHR-torus cooling (R_B) RHR-shutdown cooling (R_A)	diesel generator faults Emergency equipment cooling water	Both R_B and R_A are failed by each of these causes

less than 5% to the total. The three sequences include two types of transient initiating events and three system failures, two of which (R_B and R_A) occur only in combination. Table II-7 lists the systems and identifies the major failure contributions.

C. Peach Bottom Unit 2 Dominant Accident Sequences

Only two dominant accident sequences contribute about 75% to the total core melt probability based on the Reactor Safety Study results for Peach Bottom[3]. No other single sequence contributes more than 3%. The two sequences are T_W and T_C which are transient events followed by residual core heat (W) and scram failure (C). The residual core heat removal failure is dominated by human error and common mode failures associated with test and maintenance.

D. Dominant Accident Sequences from the Precursor Study

In June 1982, the NRC published a status report[12] describing the results to date from their Precursor Study. This study, conducted by the Oak Ridge National Laboratory and their contractor, Science Applications, Inc., uses Licensee Event Reports (LERs) to evaluate the potential for severe accident precursor events. The program involves evaluating LERs, which are required by the NRC to be prepared by plant operating utilities for various specified off-normal and unusual events, to determine which events have the potential for leading to severe core damage accident sequences. Event trees are constructed around those LERs which are determined to be significant. The event trees identify paths which, given the LER event, would lead to severe core damage. The paths represent combinations of system failures and, in some instances, initiating events. Probabilities are assigned to these failures and events to establish a quantitative estimate of core damage probability. In some cases, the LER event is not an accident-initiating event but rather the discovery that a system important to safety was in a failed or inoperable state. In these cases, the probability of an initiating event for which the unavailable system was an important mitigating factor was estimated and the core damage probability computed.

The Reference 12 status report covers LERs compiled during the 10-year period 1969 to 1979. Some 19,500 LERs were screened. It was ultimately determined that only 54 of these LER events were potentially important precursors to severe core damage accidents.

The 54 precursor events with their associated event trees and computed core damage probabilities were examined as part of this study to determine if any Oyster Creek or Millstone Unit I precursors were found which could influence the risk significance of systems and components found, or omitted from, the Millstone PRA. Of the 54 precursor events, seven involved the Millstone or Oyster Creek plants. These seven events, along with the additional failures and/or initiating events required to produce severe core damage, are listed in Table II-8. The table (last column) also provides the core damage probability as estimated from the precursor study. These probability values are arranged in decreasing order. In some cases, two different sequences involving the LER event were found to be important core damage contributors. In these cases, both sequences are listed separately as shown in the third column, and the probability value includes the total contribution from all sequences associated with the particular LER event.

As shown by Table II-8, the highest probability sequence contributes about 6×10^{-5} to the annual core damage probability at Oyster Creek, involving a loss of feedwater event. This represents about 20% of the total CMP from the Millstone PRA (assumed to apply to Oyster Creek). None of the remaining sequence probabilities in Table II-8 are greater than 2% of the total Millstone CMP.

The particular loss of feedwater event at Oyster Creek and the associated system failures were not found to be a dominant sequence in the Millstone PRA. However, all of the systems involved in the sequence (third column of Table II-8, including feedwater coolant injection, isolation condenser, and automatic depressurization system) were found to be risk significant systems and were used in this study to evaluate the SEP issues. Thus, the precursor study results do not indicate that any systems in addition to those identified in the PRAs used in this study are risk significant.

It should be noted that the precursor study report[12] has received some criticisms from various reviewers, most notably from the Institute of Nuclear Power Operations[13]. The results are, therefore, subject to controversy and should be used with caution.

Table II-8

DOMINANT CORE DAMAGE ACCIDENT SEQUENCES FOR MILLSTONE I AND OYSTER CREEK FROM ACCIDENT PRECURSOR STUDY

Plant	Event	System Failures	Probability (per yr)
Oyster Creek	Loss of Feedwater	Feedwater coolant injection, isolation condenser, and automatic depressurization system	6E-5
Millstone I	Loss of Offsite Power	1. Diesel generator and feedwater coolant injection 2. Feedwater coolant injection and automatic depressurization or low pressure injection	4E-6
Oyster Creek	Valve Malfunctions	1. Feedwater coolant injection and automatic depressurization system or low pressure coolant injection 2. Long term cooling	5E-7
Oyster Creek	Loss of Offsite Power	1. Diesel generator failures and isolation condenser 2. Feedwater coolant injection and isolation condenser and automatic depressurization system or low pressure coolant injection	5E-7
Millstone I	Diesel Generator Inoperable	Offsite power and feedwater coolant injection and isolation condenser	2E-8
Millstone I	Gas Turbine Inoperable	1. Offsite power and diesel generator and isolation condenser 2. Offsite power and isolation condenser and automatic depressurization system or low pressure coolant injection	1E-8
Millstone I	Gas Turbine Failure	1. Offsite power and diesel generator and isolation condenser 2. Offsite power and isolation condenser and automatic depressurization system or low pressure coolant injection	1E-8

III. DESIGN COMPARISON FOR BWRs USED IN THE STUDY

A brief comparison of the major design features of the Oyster Creek, Millstone[1], Browns Ferry[2], and Peach Bottom[3] BWRs was undertaken. The purpose of the comparison was to obtain some preliminary indication of the extent that the PRA plants compare with Oyster Creek. Of particular interest is the comparison between Oyster Creek and Millstone Point Unit 1 since the M.P.(1) PRA was used extensively by the NRC in their SEP evaluation[7] and also was the primary emphasis in this study. The comparison is important to help establish the extent to which the PRA results may apply to Oyster Creek.

Table III-1 illustrates the differences among the plants with respect to some 12 systems important to safety. The primary source for this comparison is Reference 8 with supplemental information from References 1, 2, 3, 9, 10 and 11. The table clearly shows that the Millstone design features are much more similar to Oyster Creek than either Browns Ferry or Peach Bottom. An inconsistency was found in the literature with respect to the existence of a safety-related feedwater coolant injection system (FWCI) at Oyster Creek. Reference 8 indicates that no such system exists at Oyster Creek, and the FSAR for the plant[11] does not include any mention of such a system. However, both References 9 and 10 indicate that the Oyster Creek design includes an FWCI system which can supply core cooling under high pressure accident conditions.

In checking further, it was determined that the Oyster Creek apparently does not have a safety grade FWCI system[14,15]. Accordingly, the Millstone core melt probabilities were adjusted to account for the lack of an FWCI system at Oyster Creek. This was done by examining each important core melt accident sequence (summarized in Table II-4 of this report from Table 8.3 of Ref. 1) and determining in which cases the FWCI failure had an important contribution to reducing the sequence probability. The FWCI function occurs in a large number of the dominant sequences for Millstone. However, in many cases, failure of the system was caused by preceding failures of support systems (predominately electric power). It was eventually determined that only three accident sequences would be appreciably affected by the absence of an FWCI system at Oyster Creek. These three sequences are designated by T_4JCD, T_4KCD, and T_3LCD in Table II-4. The probabilities of these three sequences were requantified by removing the FWCI failure probability multiplier, determined

Table III-1

COMPARISON OF BWRs USED IN STUDY

BWR	Commercial Operation	Type	Containment	RCIC	Emergency Condenser	HPCI	LPCS	LPCI	RHR	Shutdown Cooling	Containment Spray/Supp. Pool Cool.	LPCI Loop Select.	Feedwater Pump	ADS & Pressure Relief Valves	Feedwater Injec.
Oyster Creek	12/69	2	Mark I		X		X			X	X		Motor	X	
Millstone Point (1)	12/70	3	Mark I		X		X	X		X	X	X	Motor	X	X
Peach Bottom (2)	7/74	4	Mark I	X		X	X		X		X		Turbine	X	
Browns Ferry (1)	8/74	4	Mark I	X		X	X		X		X		Turbine	X	

(1) See related text, Section III.

by summing all contributions to FWCI failure as provided in Section 8 of Reference 1 for each sequence. The contribution to each release category of these sequences was recalculated using Table 8.3 of Reference 1, and the total core melt probability for each release category was summed. The result is shown in Table III-2.

Table III-2

COMPARISON OF MILLSTONE 1 AND OYSTER CREEK
CORE MELT PROBABILITY FOR EACH RELEASE CATEGORY

Plant	Release Category Probability				Total Core Melt Probability
	1	2	3	4	
Millstone 1	1E-6	8E-6	1E-4	2E-4	3E-4
Oyster Creek	1E-6	8E-6	1.2E-4	3.2E-4	4.4E-4

Aside from the feedwater injection system, Table III-1 indicates only two areas where differences exist between Oyster Creek and Millstone. The first difference is that Millstone has a low pressure coolant injection (LPCI) system while Oyster Creek does not. However, this difference is not expected to have a significant effect on the application of the Millstone risk dominant sequences to Oyster Creek since failure of LPCI for Millstone sequences always occurs concurrently with feedwater injection and LPCS failure. The failure cause for these three systems is loss of onsite emergency power rather than independent failures of the systems. Thus, it is not likely that the probability of these sequences would be different if the LPCI system were omitted.

The final difference from Table III-1 is the inclusion of an LPCI loop selection feature on Millstone and not Oyster Creek. This feature allows LPCI flow to be isolated to the recirculation loop which is postulated to rupture during a loss of coolant accident (LOCA). This feature does not improve the reliability of LPCI, but rather enhances the effectiveness of it by not allowing water to spill directly into the drywell. Further, no LOCAs of any type were found to be risk significant accident initiators. Thus, this difference is not expected to compromise the application of risk dominant sequences from the Millstone PRA to Oyster Creek.

A further major difference between Oyster Creek and Millstone was found which is not on Table III-1. Millstone Unit 1 relies on one diesel generator and one gas turbine generator for onsite emergency ac power, while Oyster Creek employs two diesel generators for this function. However, the Millstone PRA used the same failure probability for both gas turbine and diesel generators. Thus, this difference appears to be of little consequence in the context of this study.

It should be emphasized that this plant comparison study was far from exhaustive, and that risk significant differences could exist among the plants. Further, it is possible that errors and omissions exist in the PRA studies which could invalidate their results. Consequently, the risk significant evaluation of the 87 issues in the following section should be considered a preliminary scoping effort to identify any significant items.

IV. RISK SIGNIFICANT ASSESSMENT OF SEP ISSUES

Each of the 87 SEP issues as identified in the SEP report[7] was evaluated against the risk dominant systems, components, and failure causes as derived in Section II. The basic purpose of the evaluation was to determine if the issue would have any influence in terms of reducing risks. The backfit requirement imposed by the NRC for many of the issues was also examined to determine if the issues with risk reduction potential were among those with backfit requirements.

In reviewing the 87 issues, all were found to fall in 5 categories with respect to their assessed potential for risk reduction. These five categories are as follows, with definitions:

1. Undetermined - It could not be determined if risk reduction potential existed for the issue because the area being addressed was not considered in the PRA.

2. No - It could be determined with some confidence that the issue had no risk significance based on PRA results.

3. Probably Not - The potential for risk reduction appeared to be very low for the issue.

4. Possibly - The issue appeared to have some potential for risk reduction, but a definitive conclusion could not be reached for a variety of reasons.

5. Yes - The issue definitely addressed areas where risk reduction could be achieved.

Table IV-1 provides the results of the evaluation of the 87 issues. The first column provides each topic considered in the SEP, and the second column lists each of the issues (a total of 87, numbered sequentially) associated with the topic. The third column indicates whether NRC requires a backfit associated with the issue, and the fourth column provides a judgment of whether the issue appears risk significant. Comments are provided in the last column.

Table IV-2 provides a summary of the results from Table IV-1. As the table indicates, for over 40% of the issues, the risk significance was undetermined. In all of these cases, the issue was associated with external events which could pose a threat to normal operation of the plant. The PRA studies did not consider these events. For almost 50% of the issues, a determination of "no" or "probably not" regarding potential risk significance was made. Of the remaining 10 issues, only one was considered to have definite risk reduction potential. In 9 of the 10 issues, backfit requirements have been imposed. The single case of a "possibly" risk significant issue where no backfit is required involves the reactivity control issue. This issue could influence the reliability of the reactor shutdown function which would have a risk significant impact based on the PRA studies.

V. CONCLUSIONS

This section provides conclusions from the risk significance review of the 87 SEP issues, as follows:

1. The Oyster Creek core melt probability (4.4E-4 per Section IIA) as estimated from revising the results of the Millstone Point Unit 1 PRA is relatively high, as is the frequency of major release probability when compared with other PRA results. These results should be viewed with caution, however, due to the inherent uncertainties in PRA results as well as variations in rigor and methodology employed by the PRA studies and uncertainties in extrapolating the Millstone 1 results to Oyster Creek. An estimate of risk from Oyster Creek due to core melt accidents is provided in Ref. 16.

2. Of the 87 SEP issues, some judgment of potential risk significance could be made on only 52. The remainder were related to external events not considered in the PRAs. Only 10 of the 52 were found to have potential risk significance, and only one had definite risk significance. Of the 10, backfit requirements are being imposed on 9.

Table IV-1

ASSESSMENT OF 87 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
II-3.B - Flooding Potential and Protection Requirements II-3.B.1 - Capability of Operating Plants to Cope with Design Basis Flooding Conditions II-3.C - Safety Related Water Supply (UHS)	1. Condensate Transfer Pump Power [4.1(1)]	No[1]	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
	2. Flooding Level Procedures [4.1(2)]	No	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
	3. Canal Water Level Instrumentation [4.1(3)]	Yes	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
	4. Isolation Condenser Flooding [4.1(4)]	Yes	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
	5. Low Water Level Shutdown (4.1(5)]	No	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
	6. Hurricane Flooding of Pumps [4.1(6)]	Yes	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
	7. Flooding Elevation [4.1(7)]	Yes	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
	8. Groundwater Elevation [4.1(8)]	No[1]	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
	9. Roof Drains [4.1(9)]	Yes	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
III-1 - Classification of Structures, Components and Systems (Seismic and Quality)	10. Classification of Structures, Components and Systems [4.2]	Yes	Undetermined	Some components affected could have risk significance, but influence on reliability of this topis is unknown.

Table IV-1 (Continued)

ASSESSMENT OF 87 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
III-2 - Wind and Tornado Loading	11. Reactor Building Steel Structure Above the Operating Floor [4.3.1]	Yes	Undetermined	External events not considered in PRA for Millstone or Browns Ferry
	12. Ventilation Stack [4.3.2]	Yes	Undetermined	
	13. Effects of Failure of Nonseismic Category I Structures [4.3.3]	Yes	Undetermined	
	14. Components Not Enclosed in Qualified Structures]4.3.4]	No	Undetermined	
	15. Exterior Masonry Walls [4.3.5]	No	Undetermined	
	16. Roof Decks [4.3.6]	Yes	Undetermined	
	17. Intake Structure, Oil Tanks, and Diesel Generator Building [4.3.7]	Yes	Undetermined	
	18. Load Combinations [4.3.8]	No[1]	Undetermined	
	19. Soil and Foundation Capacities [4.3.9]	No	Undetermined	
III-3.A - Effects of High Water Level on Structures	20. Hydrostatic Loads (Combination) [4.4(1)]	No	Undetermined	External events not evaluated in PRA for Millstone Pt. or Browns Fer
	21. Hydrostatic Loads (Short Duration) [4.4(2)]	Yes	Undetermined	
	22. Below Grade Penetration Flooding [4.4(3)]	No	Undetermined	

Table IV-1 (Continued)

ASSESSMENT OF 87 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
III-3.C - Inservice Inspection of Water Control Structures	23. Intake and Discharge Canals [4.5.1]	No	No	
	24. Intake Structure Trash Racks and Intake Screens [4.5.2]	Yes	No	
	25. Roof Drains [4.5.3]	No(1)	No	
	26. Inspection Program [4.5.4]	No	No	
III-4.A - Tornado Missiles	27. Emergency Diesel Generators and Fuel Oil Day Tank [4.6.1]	Yes	Undetermined	External events not evaluated in PRA for Millstone Point or Browns Ferry
	28. Mechanical Equipment Access Area [4.6.2]	Yes	Undetermined	
	29. Control Room, Reactor Building, and Turbine Building Heating, Ventilating, and Air Conditioning (HVAC) Systems [4.6.3]	No	Undetermined	
	30. Condensate Storage Tank, Torus Water Storage Tank, and Service Water and Emergency Service Water Pumps [4.6.4]	Yes	Undetermined	
III-4.B - Turbine Missiles	31. Turbine Missiles [4.7]	Yes	Undetermined	Not considered in PRA (Classified as an external event)
III-4.D - Site Proximity Missiles (Including Aircraft)	32. Truck Explosions [4.8.1]	No	Undetermined	External events not evaluated in PRA for Millstone Point or Browns Ferry
	33. Aircraft Hazards [4.8.2]	Yes	Undetermined	

Table IV-1 (Continued)

ASSESSMENT OF 87 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
III-5.A - Effects of Pipe Break on Structures, Systems, and Components Inside Containment	34. Cascading Pipe Breaks [4.9(1)]	No(1)	No	Not abvious these items were definite considered in PRA studies
	35. Jet Impingement Effects [4.9(2)]	No	No	Not obvious these items were definite considered in PRA studies
	36. Drywell Penetration [4.9(3)]	No	No	Not obvious these items were definite considered in PRA studies
III-5.B - Pipe Break Outside Containment	37. LOCA Outside Containment [4.10(1)]	No	No	Neither of these events were found to be risk significant accident initiators in PRAs.
	38. Emergency Condenser Isolation [4.10(2)]	Yes	No	
III-6 - Seismic Design Considerations	39. Pipe Systems [4.11(1)]	Yes	Undetermined	Seismic risks not considered in Millstone or Browns Ferry PRAs. See related discussion in text.
	40. Mechanical Equipment [4.11(2)]	Yes	Undetermined	
	41. Electrical Equipment [4.11(3)]	Yes	Undetermined	
	42. Ability of Safety-Related Electrical Equipment to Function [4.11(4)]	No	Undetermined	
	43. Qualification of Cable Trays [4.11(5)]	Yes	Undetermined	
III-7.B - Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria	44. Design Codes, Design Criteria, Load Combinations and Reactor Cavity Design Criteria [4.12]	Yes	Undetermined	Loads being considered are principally from external events not considered in reference PRAs. Influence of codes and criteria not addressed in PRAs.

Table IV-1 (Continued)

ASSESSMENT OF 87 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
III-8.A - Loose-Parts Monitoring and Core Barrel Vibration Monitoring	45. Loose-Parts Monitoring and Core Barrel Vibration Monitoring [4.13]	No	No	No accident initiating events of risk significance would be prevented by this provision based on PRAs.
III-10.A - Thermal-Overload Protection for Motors of Motor-Operated Valves	46. Thermal Overload Bypass [4.14(1)]	Yes	Probably Not	Valves appear as contributors to two risk significant systems (Isolation condenser makeup and residual heat removal). However, valve failure report (NUREG/CR-1363) shows no failures attributed to malfunction in thermal overload protection, although a large fraction (24%) of unknown failures are listed.
	47. Magnetic Trip Breakers [4.14(2)]	No	Probably Not	
IV-2 - Reactivity Control Systems Including Functional Design and Protection Against Single Failures	48. Reactivity Control Systems, Including Functional Design and Protection Against Single Failures [4.15]	No	Possibly	Scram failure is a risk significant event. Not clear if the requirement involves effort that would reduce scram failure probability, which was determined to be common mode mechanical failure of control rods to insert for Millstone Point.
V-5 - Reactor Coolant Pressure Boundary (RCPB) Leakage Detection	49. Leakage Detection Systems [4.16.1]	Yes	No	Pipe failure LOCAs not risk significant and leakage detection systems not accounted for in PRAs
	50. Operability Requirements [4.16.2]	Yes	No	
	51. Intersystem Leakage [4.16.3]	No	No	Intersystem leakage not risk significant from PRAs
	52. Reactor Coolant Inventory Balances [4.16.4]	No	No	

Table IV-1 (Continued)

ASSESSMENT OF 87 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
V-6 - Reactor Vessel Integrity	53. Reactor Vessel Integrity [4.17]	Yes	No	Reactor vessel failure not risk significant from PRAs.
V-10.B - Residual Heat Removal System Reliability	54. Residual Heat Removal System Reliability [4.18]	Yes	Yes	RHR is a risk significant system and it appears from discussion on Page 4-26,27 that reliability would be improved.
V-11.A - Requirements for Isolation of High- and Low-Pressure Systems	55. Requirements for Isolation of High- and Low-Pressure Systems [4.19]	Yes	No	RWCU is only system affected, and it is not a risk significant system failure based on PRAs.
V-12.A - Water Purity of BWR Primary Coolant	56. Water Purity of BWR Primary Coolant [4.20]	Yes	No	
VI-1 - Organic Materials and Post-accident Chemistry	57. Organic Materials [4.21.1]	Yes	No	
	58. Post-Accident Chemistry [4.21.2]	No	No	
VI-4 - Containment Isolation System	59. Locked-Closed Valves [4.22.1]	Yes	No	See discussion of containment isolation in text.
	60. Remote Manual Valves [4.22.2]	Yes	No	
	61. Valve Location [4.22.3]	No	No	
	62. Instrument Lines [4.22.4]	No	No	
	63. Valve Location and Type [4.22.5]	No	No	
	64. Administrative Controls [4.22.6]	No	No	

Table IV-1 (Continued)

ASSESSMENT OF B7 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
VI-7.A.3 - Emergency Core Cooling Actuation System	65. Emergency Core Cooling System Actuation System [4.23]	Yes	No	Emergency condenser failure not risk significant (only condenser makeup)
VI-7.A.4 - Core Spray Nozzle Effectiveness	66. Core Spray Nozzle Effectiveness [4.24]	No	No	No low pressure accidents are risk significant. Also, core spray effectiveness not considered in PRA studies.
VI-7.C.1 - Appendix K - Electrical Instrumentation and Control Re-Reviews	67. AC Automatic Bus Transfers [4.25(1)]	Yes	Possibly	Loading of emergency ac buses is risk significant based on Millstone Point PRA; influence of automatic bus transfers not evaluated.
	68. DC Automatic Bus Transfers [4.25(2)]			
VI-10.A - Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing	69. Response-Time Testing [4.26.1]	No	No	Mechanical failure of control rods dominates system failure
	70. Instrumentation for Reactor Trip System (RTS) Testing [4.26.2]	Yes	Possibly	Scram failure is risk significant. The influence of this issue on scram reliability at Oyster Creek has not been evaluated.
	71. Dual-Channel Testing [4.26.3]	No	Probably Not	Based on evaluation on 4-37 of SEP (not significant for Millstone, but not included in Table 4.1)

Table IV-1 (Continued)

ASSESSMENT OF 87 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
VII-1.A - Isolation of Reactor Protection System from Nonsafety Systems, Including Qualifications of Isolation Devices	72. Flux Monitoring Isolation [4.27(1)]	Yes	Possibly	See Page 4-38 of SEP
	73. Reactor Protection System (RPS) Protective Trip [4.27(2)]	Yes	Possibly	
VII-1.B - Trip Uncertainty and Setpoint Analysis Review of Operating Data Base	74. Trip Uncertainty and Setpoint Analysis Review of Operating Data Base [4.28]	Yes	Possibly	
VII-2 - Engineered Safety Features System Control Logic and Design	75. Engineered Safety Features System Control Logic and Design [4.29]	No(1)	Probably Not	See #47 [4.14(1)], the only item found by the SEP to be in non-compliance for Oyster Creek
VII-3 - Systems Required for Safe Shutdown	76. Systems Required for Safe Shutdown [4.30]	Yes	Possibly	
VIII-2 - Onsite Emergency Power Systems (Diesel Generators)	77. Onsite Emergency Power Systems (Diesel Generator) [4.31]	Yes	Probably Not	Problems found at Oyster Creek diesels relative to this topic are not significant contributions to failure (see related text).
VIII-3.B - DC Power System Bus Voltage Monitoring and Annunciation	78. DC Power System Bus Voltage Monitoring and Annunciation [4.32]	Yes	Probably Not	DC bus failure not found to be risk significant in Millstone Point and Browns Ferry PRAs. Issue does not appear to have a large influence on DC availability at Oyster Creek (Ref. pages 4-42 and 111, App. D of SEP report).

Table IV-1 (Continued)

ASSESSMENT OF 87 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
VIII-4 - Electrical Penetrations of Reactor Containment	79. Electrical Penetrations of Reactor Containment [4.33]	No	No	Penetration failure not significant to risk based on PRAs. See related discussion in text.
IX-5 - Ventilation Systems	80. Restoration of Ventilation [4.34(1)]	Yes	Possibly	Could influence diesel generator reliability
	81. Reactor Building Ventilation [4.34(2)]	No	No	
	82. Core Spray and Containment Spray Pump Ventilation [4.34(3)]	Yes	No	These systems not important to risk based on PRAs for Millstone and Browns Ferry. Does not appear that problem at Oyster Creek will decrease reliability to the extent that systems become risk significant.
	83. Battery, Motor-Generator and Switch Gear Ventilation [4.34(4)]	Yes	Possibly	Affected equipment may influence reliability of risk significant system.
XV-1 - Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	84. Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve [4.35]	No	Probably Not	These problems could influence the frequency of transient initiating events for risk significant sequences. However, experience at Oyster Creek (Appendix F, pg. F-xii) shows apparently very little influence from these problems on the frequency of transients, which is not unusual compared to data for other plants.

Table IV-1 (Continued)

ASSESSMENT OF 87 ISSUES AGAINST RISK SIGNIFICANT SYSTEMS

Topic	Issue	Backfit Required	Risk Significant	Comments
XV-16 - Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment.	85. Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment [4.36]	Yes	No	Not found to be risk significant in PRAs.
XV-18 - Radiological Consequences of a Main Steam Line Failure Outside Containment	86. Radiological Consequences of a Main Steam Line Failure Outside Containment [4.37]	No(1)	No	Not found to be risk significant in PRAs.
XV-19 - Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Pipe Breaks Within the Reactor Coolant Pressure Boundary	87. Loss-of-Coolant Accident Resulting from Spectrum of Postulated Pipe Breaks Within the Reactor Coolant Pressure Boundary [4.38]	Yes	No	Not found to be risk significant in PRAs.

(1) According to the SEP, these issues are partially covered by other backfit requirements (See Table 4.1 of SEP).

Table IV-2

SUMMARY OF ISSUE EVALUATIONS

Risk Significant	No. of Issues	No. of Issues Requiring Backfit	Comments
Undetermined	35	22	3 others partly covered under related backfitting
No	35	14	3 others partly covered under related backfitting
Probably Not	7	3	1 other partly covered under related backfitting
Possibly	9	8	No backfit required for reactivity control issue (#48 from Table IV-1)
Yes	1	1	Reliability of RHR (#54 from Table IV-1)

The single issue not covered by backfit requirements is "Reactivity Control Systems, Including Functional Design and Protection Against Single Failures". The NRC position on this issue as described in the SEP was evaluated to determine if conclusions could be established on its risk significance.

On Page 4-22 of the SEP, the NRC concludes that, on the basis of a "limited PRA" (which apparently refers to Appendix D of the SEP), the effects of a multiple rod withdrawal on risk was determined to be of low risk. This conclusion appears valid. However, the topic would seem to cover also those situations with influence scram reliability. The NRC staff further concludes (pg. 4-22) that the Oyster Creek rod control system design, based on a review at Dresden Unit 2, meets current licensing criteria, and thus backfitting is not recommended.

On Page A-34, it is concluded for this topic that "NRC has concluded that revisions to existing licenses are not warranted. Staff effort on this issue will continue at a low level."

On Page 25 of Appendix D where Topic IV-2 is considered, under the sub-heading "NRC evaluation", the following statement appears: "There exists the potential for single failures at Oyster Creek which would allow multiple control rod withdrawals, which is considered not acceptable." This conclusion seems inconsistent with the evaluation on Page 4-22.

The assessment in Appendix D also includes the results (no details given) of a fault tree evaluation of the Oyster Creek reactor protection system, and a "careful" evaluation was made of the fault tree and the equivalent fault tree for Millstone. It was concluded from this effort that "... failures in the rod withdrawal part of the control rod system do not contribute to failure of the RPS." This is probably a valid assessment. However, it addresses a narrow interpretation of the topic and does not consider scram reliability. Further, the increased probability of uncontrolled rod withdrawal would increase the probability of transient accident initiators (designated T_2 in the Millstone PRA) requiring scram. This increase is probably not significant, however, compared to the 6.6 event/year used in the PRA.

It appears that the NRC position is somewhat vague and inconclusive. As a result, this topic is considered to remain in the "possibly" category with respect to risk significance.

3. An investigation was performed of a recent study[12] to determine if identified accident precursors at either Millstone Unit 1 or Oyster Creek appeared significant in terms of identifying SEP issues of risk significance. The study was undertaken by ORNL and involved examining actual events at nuclear power plants for the time period from 1969 to 1979. The events were screened to identify those which had significant potential for leading to core melt accidents if additional system failures had occurred. For those with significant potential, a core melt probability was calculated by identifying subsequent system failures required to produce core melt and assessing the probability of such failures given the state of the plant at the time of the event.

   The precursor study identified 54 precursor events (from Licensee Event Reports) with potential for adding to the probability of core melt. The Oyster Creek and Millstone Unit 1 reactors were involved in seven of these precursor events. However, the assessed core melt probability for these seven events did not in any case exceed 5% of the core melt probability assessed for Millstone Unit 1. Thus, it is concluded that the precursor study did not identify the potential for any core melt accident sequences for Oyster Creek or Millstone with a significant probability compared to the Millstone PRA results. Further details may be found in Section II.D.

REFERENCES

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2. Interim Reliability Evaluation Program: Analysis of the Browns Ferry Unit 1, Nuclear Plant, NUREG/CR-2802, EG&G Idaho, July 1982.

3. Reactor Safety Study, WASH-1400, USNRC, October 1975.

4. PRA Procedures Guide, NUREG/CR-2300, USNRC, January 1983.

5. Inside NRC, January 24, 1983, McGraw-Hill.

6. Overview of the Reactor Safety Study Consequences Model, NUREG-0340, USNRC, June 1977.

7. Integrated Plant Safety Assessment - Systematic Evaluation Program, Oyster Creek Nuclear Generating Station, NUREG-0822, January 1983.

8. Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, NEDO-24708, General Electric Co., August 1979.

9. Summary Report on a Survey of Light-Water-Reactor Safety Systems, NUREG/CR-2069, F. A. Heddleson, ORNL, April 1983.

10. Design Data and Safety Features of Commercial Nuclear Power Plants, ORNL-NSIC-55, December 1973.

11. Final Safety Analysis Report - Oyster Creek Nuclear Plant, Jersey Central Power & Light Co.

12. Precursors to Potential Severe Core Damage Accidents: 1969-1979 - Status Report, NUREG/CR-2497, Oak Ridge National Laboratory, June 1982.

13. Review of NRC Report: Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report, NUREG/CR 2497, INPO-82-025, September 1982.

14. Personal Communication with Mike Hober, New Jersey Bureau of Radiation Protection, June 28, 1984.

15. Grouping of Light Water Reactors for Evaluation of Decay Heat Removal Capability, NUREG/CR-3713, Brookhaven National Laboratory, June 1984.

16. An Estimate of Risk from Core Melt Accidents at the Oyster Creek Reactor (Appendix A), P. R. Davis, to be published.