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Value-Impact Analysis of Regulatory Options for Resolution of Generic Issue C-8: MSIV Leakage and LCS Failure

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ABSTRACT

This report describes the analysis conducted to establish the basis for answering two remaining regulatory questions facing the NRC staff regarding the resolution of Generic Issue C-8, specifically:

1) What action should the NRC take concerning plants that currently have a leakage control system (LCS)?

2) What action should the NRC take concerning plants that do not have an LCS?

Using individual MSIV leak test data, the performance of a system of eight such valves in a standard BWR con-figuration was modeled. The performance model was used along with estimates of core damage accident frequency and calculated dose consequences to determine the public risk associated with each of the alternatives. The occupational exposure implications of each alternative were calculated using estimates of labor hours in radiation zones that would be incurred or avoided. The costs to industry of implementing each alternative were estimated using standard cost formulae and NRC staff estimates. The costs to the NRC were estimated based on the effort incurred or avoided for reviews or other staff actions engendered by the selection of a particular alternative. The costs and risks thus calculated suggest that no regulatory action can be justified on the basis of risk reduction or cost savings.

SUMMARY

The subject of Generic Issue C-8 is the inability of some Main Steam Isolation Valves (MSIVs) in boiling water reactors (BWRs) to consistently meet the technical specification leakage rate limit. Based on conservative analyses and the desire to limit offsite doses, a Leakage Control System (LCS) has been required on some BWRs to direct any leakage past the MSIV during a Loss of Coolant Accident (LOCA) to an area served by the Standby Gas Treatment System (SGTS). Due to limitations in its design, the LCS may not be effective if the leakage rate through the MSIV is greatly in excess of the technical specification value.

During FY 85-86, Pacific Northwest Laboratory (PNL) performed a technical evaluation of Generic Issue C-8. The results of that evaluation are reported in NUREG-1169 (NRC 1986). That report describes the causes of excessive MSIV leakage as assessed by the BWR Owners' Group and the Nuclear Regulatory Commission staff, and presents PNL's evaluation of alternative leakage control methods using a realistic fission product transport model.

The objective of this value-impact analysis is to establish the basis for answering two remaining regulatory questions facing the NRC staff, specifically:

- What action should the NRC take concerning plants that currently have AN LCS?

- What action should the NRC concerning plants that do not have an LCS?

The alternatives considered were:

1) Require plants with negative-pressure LCSs to take them out of service.

2) Require plants with negative-pressure LCSs to upgrade them to higher capacity.

3) Require plants without an LCS to install a safety-grade LCS with capacity comparable to those now in service.

Using individual valve leak-rate test performance data provided by the Boiling Water Reactor Owners' Group (BWROG) the performance of a system of such valves in a standard BWR configuration (four steam lines in parallel, each with two valves in series) was modeled. The leakage model was used in conjunction with representative core damage accident frequencies from published probabilistic risk assessments (Hatch 1987, NRC 1981, NRC 1989) and dose consequences calculated using the CRAC2 computer code (Ritchie 1987) to determine the public risk from MSIV leakage associated with each of the alternatives.

The occupational exposure implications of each alternative were calculated using estimates of labor hours in radiation zones that would be incurred or avoided by implementing each of the alternatives. These changes in occupational radiation exposure were balanced against the increase or decrease in public risk to determine the net radiation risk benefit of each alternative.

Finally, the costs to the industry and of implementing each alternative were estimated using standard cost formulae and NRC staff estimates of the cost to install a new LCS at an operating plant. The cost to the NRC was quantified by estimating the effort avoided or incurred for reviews, license amendments or other staff actions engendered by the selection of a particular alternative.

The costs and risks for each alternative are summarized in the following table. The conclusion suggested by the summary is that no regulatory action can be justified based on the risk to be avoided or cost to be saved.

Summary of Costs and Benefits

	Alternative 1 (Disable LCS)	Alternative 2 (Upgrade Capacity)	Alternative 3 (Add LCS)
Radiation exposure (person-rem)[(a)]			
Public	-3.3E+2	1.4E+2	6.2E+2
Occupational	7.0E+1	-2.95E+2	5.83E+2
Total	-2.6E+2	-1.6E+2	3.7E+1
Monetary costs[(b)]			
Industry implementing costs	$1.68E+5	$6.70E+6	$1.50E+7
Industry operating costs	-$3.23E+5	$3.23E+5	$4.57E+5
NRC Costs	$2.1E+5	$4.14E+5	$4.45E+5
Total	$5.5E+4	$7.4E+6	$1.6E+7

(a) Negative exposure numbers represent increased exposure.
(b) Negative cost numbers represent cost savings.

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ACRONYMS/ABBREVIATIONS

BWR/PWR	boiling water reactor/pressurized water reactor
BWROG	Boiling Water Reactor Owners' Group
CsI	cesium iodide
DBA	design basis accident
EPRI	Electric Power Research Institute
HEPA	high efficiency particulate air filters
hr	hour
IC	isolated condenser
LCS	leakage control system
LLRT	local leak rate test
LOCA	loss of coolant accident
LOOP	loss of offsite power
MSIV	main steam line isolation valve
MSIV-LCS	main steam line isolation valve-leakage control system
MSL	main steam line
person-hr	person-hour
PNL	Pacific Northwest Laboratory
PRA	probabilistic risk assessment
psi	pounds per square inch
psia	pounds per square inch absolute
RCS	reactor coolant system
RPV	reactor pressure vessel
scfd	standard cubic feet per day
scfh	standard cubic feet per hour
scfm	standard cubic feet per minute
SGTS	standby gas treatment system
SJAE	steam jet air ejectors
SRV	safety relief valve
TBV	turbine bypass valve
TCV	turbine control valve

1.0 STATEMENT OF PROBLEM

The boiling water reactors (BWRs) widely used in this country for generation of electricity make use of two Y-pattern globe valves in series to isolate each of the main steam lines from the balance of the plant. These main steam isolation valves (MSIVs) form part of the primary containment boundary and therefore must seal tightly with high reliability to prevent uncontrolled release of radioactive material to the plant environs in the event of a core damage accident. For that reason the valves are subject to periodic leak-rate testing to ensure their continuing operability. Stringent limits on allowable leakage, typically 11.5 standard cubic feet per hour (scfh) at a test pressure of 25 psi, were established early in the history of commercial BWR operation to limit the potential radiological consequences of a major reactor accident.

During the early 1970s as the first group of large BWRs entered commercial operation, a significant number of MSIV leak test failures were seen. Because of concern that excessive MSIV leakage could compromise the function of the reactor containment and lead to large dose consequences in the event of a reactor accident, the U.S. Nuclear Regulatory Commission (NRC) issued guidance (Regulatory Guide 1.96) that had the effect of requiring that most BWRs with construction permits issued after March 1, 1970, be provided with MSIV leakage control systems (LCSs). The NRC's concern over MSIV performance increased in the late 1970s and early 1980s as more and more leak test failures were reported, some at very high leak rates. Because the gross leak rates observed at some plants were shown to have the potential for causing offsite doses exceeding the siting criteria limits in the Code of Federal Regulations (10 CFR 100), Generic Issue C-8 was created to assess the causes of MSIV leakage, the effectiveness of the LCS and other methods of mitigating accident releases, and the need for regulatory action to limit public risk.

This report documents the value-impact analysis performed by the Pacific Northwest Laboratory (PNL)[a] of the options available to the NRC for accomplishing the final resolution of Generic Issue C-8.

1.1 NRC REGULATORY BACKGROUND

The requirements for control of MSIV leakage are based on "General Design Criteria for Nuclear Power Plants," Appendix A of 10 CFR 50 (10 CFR 50). Specifically, Criterion 54 requires that:

> Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability

(a) Operated by Battelle Memorial Institute for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55 requires that:

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves--(1) one automatic isolation valve inside and one automatic isolation valve outside containment....

The requirements for "Primary Reactor Containment Leakage Testing" for water-cooled reactors are found into Appendix J, 10 CFR 50. As implemented, the Appendix J requirements typically result in MSIVs being leak tested every refueling outage by local pressurization to about 25 pounds per square inch absolute (psia) with air or nitrogen. The leak rate limit, as specified in the plant-specific technical specifications, is typically 11.5 scfh, a number that has its basis in a conservative assessment of offsite dose consequences. The dose assessment methodology is described in Standard Review Plan Sections 15.6.4 and 15.6.5, and uses the U.S. Atomic Energy Commission (AEC) TID-14844 (1962) source term assumptions and 10 CFR 100 dose guidelines as acceptance criteria.

The NRC's concern over the possible dose consequences of MSIV leakage at or above the technical specification leakage limits led to the requirement that a LCS be installed in new plants. This regulatory position was set forth in Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants." Currently, there are eleven BWR 4 and BWR 5 plants that have such a system. All BWR 6 plants have LCSs.

Two fundamentally different types of LCSs have been implemented by licensees to mitigate the effects of MSIV leakage. The first type uses a positive back-pressure of nitrogen or air in the main steam line (MSL) between the MSIVs to prevent outward leakage. The second, or more prevalent type, uses fans or "exhausters" to maintain a subatmospheric (negative) pressure in the steam lines between the MSIVs. Any leakage past the MSIVs is thereby collected, routed through a filtered exhaust system, and discharged to the environment. Details of LCS system design differ from plant to plant. The description and principles of operation presented in Section 2.3 of NUREG-1169 (NRC 1986) are typical of most systems in use today. Table B.2 in Appendix B contains a list of all operating BWRs and the type of LCS (if any) installed.

1.2 LEAKAGE OF MAIN STEAM ISOLATION VALVES AT BOILING WATER REACTORS

In the period from about 1970 to 1981, many BWR licensees reported difficulty meeting the allowable leakage rate limit for periodic local leak rate tests (LLRTs). A 1976-77 survey of the leakage rates of 400 MSIVs showed that 46 valves exceeded the leakage rate limit. A relatively large number of valves leaked far in excess of the allowable limit, with some leakage rates as high as 3795 scfh being reported. At one plant, some valves were reported to consistently have a test leak rate much greater than 11.5 scfh; some were consistently above 1000 scfh.

Another survey of MSIV performance at BWRs for the years 1979 through 1981 revealed that 18 of 25 operating BWRs had MSIVs that had excessive leak rates during one or more surveillance tests. During this time the number of MSIV test failures exceeded 150, including MSIVs supplied by all three MSIV vendors. Leak rates that exceeded the limit ranged up to 3427 scfh. From these data it was estimated that only 58% of all MSIV leak tests produced results that were within the 11.5 scfh specification. Seventeen percent produced results that were between 11.5 scfh and 100 scfh. Twenty-five percent exhibited leakage greater than 100 scfh. A mean leak rate of 1500 scfh, more than 100 times the specification, was determined for this latter group (NRC 1983).

To return the valves to within the allowable leakage rate limits, different methods of refurbishment have been used. Most utilities grind or lap the valve seating surfaces. At least one utility has instituted a major refurbishment that includes increasing the actuator stem diameter, modifying the guide rails for the valve plug, and increasing the force of the valve operator.

1.3 SUMMARY OF EFFORTS TO RESOLVE GENERIC ISSUE C-8

As a result of the safety concerns described earlier in this section, the NRC staff, with contractor assistance and the BWR Owners' Group (BWROG) initiated independent efforts to answer the technical questions surrounding the MSIV leakage issue. The BWROG formed the MSIV Leakage Control Committee to determine the cause of the high leakage rates associated with many of the MSIVs and to develop recommendations to reduce the leakage rates. The BWROG committee completed its efforts and provided recommendations to the staff in February 1984.

The efforts of the NRC staff and contractors were directed at determining: 1) the adequacy of industry efforts to identify and correct causes of excessive MSIV leakage, 2) the basis for any change in the allowable MSIV leakage rate, 3) the need for a safety-grade LCS, and 4) the specific areas of regulations and guidance that may be necessary to implement the findings. The approach was to evaluate the effects of MSIV leakage in terms of offsite doses following a loss-of-coolant accident (LOCA), using realistic assumptions concerning the equipment, facilities and site characteristics available to mitigate the effects of a LOCA. The specific elements of the effort were:

- to evaluate the BWROG recommendations associated with reducing leakage through MSIVs and assess the effectiveness of the recommendations as implemented by licensees

- to evaluate the existing safety-related LCS by comparing its effectiveness with that of other possible methods of handling leakage following a LOCA

- to perform a probabilistic risk assessment (PRA) to evaluate the reliability and relative risks associated with the different methods of mitigating the effects of a LOCA
- to evaluate the use of alternate equipment to mitigate the effects of a LOCA and potential changes in allowable MSIV leakage rates
- to identify candidate areas for changes in current licensing guidance, including the Regulations, the Standard Review Plan, Regulatory Guides, and Technical Specifications.

The results of that work are documented in Resolution of Generic Issue C-8, NUREG-1169 (NRC 1986).

2.0 OBJECTIVES IN RESOLVING GENERIC ISSUE C-8

As stated in the NRC staff's May 1983 Task Action Plan for Generic Issue C-8, the objectives of the resolution were to:

- determine the reasons for high MSIV leakage rates and identify areas where corrective action could be taken
- evaluate the benefit of a safety-related leakage control system and determine the optimum means of handling MSIV leakage under credible LOCA conditions
- evaluate the technical specification leak rate limit to determine if there is a basis for any change
- identify appropriate changes to Regulatory Guides, Standard Review Plans, and Technical Specifications.

In this section, each of these original objectives are discussed in turn, and the evolution of the objectives of this analysis are traced. Finally, the questions guiding this value-impact analysis are also stated.

2.1 DETERMINE REASONS FOR MSIV LEAKAGE AND IDENTIFY CORRECTIVE ACTIONS

The leakage control committee of the BWR Owners' Group undertook a program of data collection and evaluation directed at identifying the causes of high MSIV leakage. The committee provided the results of its work to the NRC in February 1984 and many BWR licensees proceeded to implement them. The committee's recommendations were reviewed by PNL. It was determined that: 1) the committee had done a credible job of data collection and analysis concerning the causes of MSIV leakage and leak test failures, 2) there was a high likelihood that the key causes of excessive MSIV leakage had been identified, and 3) the committee's recommendations, if implemented, would probably solve most of the leakage problem. The results of the PNL review were published in Appendix A of NUREG-1169 (NRC 1986).

At the time of the publication of NUREG-1169, no data were available to characterize MSIV leak test performance in industry. Over the period 1984-87, the MSIV Leakage Control Committee's recommendations on maintenance practices, modifications and leak test methods were disseminated by the Owners' Group and implemented by many licensees. In 1987 and early 1988, data were collected from the Owners' Group members that indicated a marked improvement in valve performance. The data was formally transmitted to the NRC in August 1988 and is presented in Appendix A of this report. Whereas the record of the 1970-81 period was one of frequent leak-test failures (sometimes more than half of any given group of valves) and large leak rates, the 1987-88 data show that over 75% of the valves tested met the 11.5-scfh specification. More significantly, no instances of leakage greater than 1000 scfh were reported from 329 tests. In fact, only five of the 329 (1.5%) tests yielded leak rates in excess of 100 scfh and only one exceeded

500 scfh. This contrasts sharply with the 1979-81 survey results cited in NUREG-0933 (NRC 1983) in which 25% of all leak test results exceeded 100 scfh; the median for that group was 1500 scfh. The BWROG data base included test results from 24 of 30 plants represented on the leakage control committee, or about 63% of the 38 BWR plants now in commercial operation. It included an approximately proportional representation of valves from all three major valve manufacturers. Because the leakage control committee included representatives from most of the plants that historically had the worst leak-test failure problems, the data base was judged to conservatively represent the capabilities of the whole BWR plant population. The data are interpreted to mean that industry has identified the major causes of MSIV leakage and successfully corrected them.

2.2 EVALUATE THE BENEFIT OF A SAFETY-RELATED LEAKAGE CONTROL SYSTEM

The industry experience with leaking MSIVs raised several questions about the usefulness of the safety-grade LCSs that were installed in some operating BWRs. These LCSs are typically designed to handle flows only moderately in excess of the MSIV leakage specification. If total leakage for all steam lines is greater than about 400 scfh, the "typical" LCS becomes nonfunctional. Thus, for the MSIV leakage events of most safety concern (in the thousands of scfh), the LCS would not be operable. Also, there are about 20 BWRs operating without leakage control systems.

Even for leakage rates within the capacity of the LCS, there were concerns that operating the system might actually result in <u>higher</u> offsite doses in the event of a core damage accident by expediting the release to the environment of large amounts of short-lived noble gas activity that otherwise might decay while still contained inside plant systems and piping.

Because of these factors, Generic Issue C-8 was formulated to answer the following questions:

- What other (non-safety grade) systems and methods would likely be available to mitigate the offsite dose consequences of MSIV leakage?

- How do these alternate methods compare with the "standard" LCS in terms of cost for implementation, maintenance, availability, and public risk reduction?

The evaluation concluded that several methods existed by which MSIV leakage far in excess of the LCS capacity could be attenuated by use of the main steam lines, condenser shell and main turbines as holdup volumes. These components and the structures that house them are typically nonseismic Category I, and credit cannot usually be taken for them in licensing analyses intended to show compliance of a plant design with NRC regulations. However, the modeling and analysis documented in NUREG-1169 (NRC 1986) established the following facts:

1. There are several methods by which BWR licensees could use non-safety grade equipment systems and components to mitigate the effects of high MSIV leakage.

2. The alternate treatment methods can be more effective than the LCS in reducing public risk following a core damage accident, particularly for those cases when the total MSIV leakage rate exceeds LCS capacity.

3. The alternate treatment methods generally have high availability and costless to implement.

The key structures, systems and components that would be used in these alternate methods are typically not designed to seismic Category I criteria. However, there is substantial earthquake response data (ASCE 1986) indicating that valves, steam lines, condensers and the buildings that house them frequently survive earthquake accelerations far exceeding their design bases without loss of structural integrity, and in many cases, without any functional impairment.

2.3 EVALUATE TECHNICAL SPECIFICATION LEAK RATE LIMIT

The demonstrated inability of MSIVs to consistently meet the leak rate limit indicated that the 11.5-scfh specification might be unnecessarily stringent and, in fact, unrealistic for such large valves. However, analyses documented in NUREG-1169 (NRC 1986) and by Niagara Mohawk Power[a] indicated that a large general increase in the leak rate specification was not likely to be justifiable for the following reasons:

1. Any increase in allowable MSIV leak must still be justified by showing that the resulting offsite doses in the event of a design basis LOCA do not exceed 10 CFR 100 criteria.

2. In a practical sense, the control room dose from MSIV leakage may preclude large increases for some licensees because the cost of upgrading the control room habitability would likely exceed any savings realized from reduced valve maintenance.

3. The effect on MSIV performance of maintaining the valves to a less stringent specification cannot be quantified at this time. The valve performance data used in this value-impact analysis were the result of concerted industry efforts to meet the current limit (11.5 scfh). A large increase in the limit could have a broad negative impact on the industry test and maintenance standards. Also, it could result in a downturn in the valve performance upon which current regulatory decisions are now being made.

(a) Niagara Mohawk Power Corp. Letter to U.S. Nuclear Regulatory Commission dated March 18, 1987. (NMP21 1007) Subject: Nine Mile Point Unit 2, Docket No. 50-410.

4. Because of industry's improvements in valve maintenance and testing methods in the past few years, most valves routinely pass the test and few, if any, incidents of gross leakage are reported. This indicates that the current limit is not unrealistic or unachievable.

2.4 IDENTIFY CHANGES TO REGULATORY GUIDES, STANDARD REVIEW PLANS, AND TECHNICAL SPECIFICATIONS

The parts of applicable Regulatory Guides, Standard Review Plans, and Technical Specifications that were potential candidates for revision to reflect the results of the Generic Issue C-8 resolution effort are outlined in Section 7 of NUREG-1169 (NRC 1986).

2.5 SPECIFIC OBJECTIVES OF THIS VALUE-IMPACT ANALYSIS

This analysis is intended to provide a basis for answering two remaining regulatory questions facing the NRC staff:

1. What action should the NRC take concerning plants that currently have an LCS?

2. What action should the NRC take concerning plants that do not have an LCS?

The alternatives considered for each of these questions will be presented and analyzed in Section 3.0.

3.0 ALTERNATIVES

The alternatives considered here are of two types: those that apply to plants with a negative pressure LCS and those that apply to plants that have no LCS.

Of the 14 plants (average remaining life of 34 years) that currently have a negative pressure LCS, the following alternatives are considered:

1. Require plants with negative pressure LCSs to take them out of service by closing isolation valves and disconnecting controls such that they cannot be inadvertently initiated. Plant procedures and technical specifications would be modified to eliminate use and required surveillance of the LCS.

2. Require plants with negative pressure LCSs to upgrade them to higher capacity (~4000 scfh).

Of the 21 plants (average remaining life of 25 years) that currently do not have an LCS, the following alternative is considered:

3. Require plants without an LCS to install a safety-grade LCS with capacity comparable to those now in service (~400 scfh).

4.0 CONSEQUENCES

The alternatives concerning the present situation are compared here. The costs and impacts of each alternative are quantified and balanced against the benefits of that alternative.

4.1 BENEFITS AND COSTS

The quantifiable values to be examined here include changes in the following:

- public risk due to MSIV leakage
- occupational exposure
- industry implementing costs
- industry operating costs
- NRC costs.

Two other frequently considered cost elements, public property damage, and onsite property damage are responsive only to changes in core damage frequency and, therefore, are unaffected by any of the alternatives being examined here.

4.1.1 Public Risk Due to MSIV Leakage

None of the proposed alternatives would alter the core damage frequency per plant-year. However, by changing the amount of radioactive material emanating from a plant following a core damage accident, implementation of an alternative could impact the population dose, and hence public risk from some or all plants.

As shown in Appendix B, public risk for all core damage causes, MSIV leak rates, and release pathways was calculated using the CRAC2 computer code (Ritchie 1983), the MSIV leakage model developed in Appendix C, and representative core damage accident frequencies from published probabilistic risk assessments (Hatch 1987, NRC 1981, NRC 1989). The current public risk for all core damage causes, leak rates, and release pathways is 1.1E+3 person-rem. Each of the alternatives is assumed to cause a change in the magnitude or type of radioactive release, and hence, the public risk that would result. This change will be presented as the decrease (or increase) in public risk that would result from implementation of the alternative.

Alternative 1: Require plants with negative-pressure LCSs to take them out of service.

This option does not alter core damage frequency. It does result in a larger release for those core damage accidents with total MSIV leakage that would otherwise be within the capacity of the LCS.

The public risk for this option is calculated by assuming that all MSIV leakage is released by the least effective alternate pathway (isolated steam line) for the accident cases when the steam lines would be assumed intact (internal and non-seismic event-induced core damage accidents). For seismically induced core damage accidents, the steam lines are assumed to fail and the release is assumed to be direct to the atmosphere without benefit of holdup in the steam lines. The total risk can be represented as the sum of the risk from internal and non-seismic induced core damage accidents and that from seismic-induced core damage accidents. From Appendix B, this is shown to be 3.3E+2 person-rem for the BWR population now equipped with LCSs.

The public risk from the existing situation is shown in Appendix B to be 1.1E+3 person-rem. The net effect on public risk of implementing Alternative 1 is, therefore, (1.1E+3) + (3.3E+2) or 1.4E+3 person-rem, or an increase of 330 person-rem over the existing situation.

Alternative 2: Require plants with LCSs to upgrade the systems to higher capacity (~4000 scfh).

This alternative does not alter core damage frequency but would provide greater than 99.9% assurance that seismic Category I LCSs would have sufficient capacity to handle the combined leakage from all MSIVs. In Appendix B it is shown that the public risk from implementation of this alternative for plants with LCSs is 1.7E+1 person-rem compared with 1.6E+2 person-rem for the existing situation. The net effect of implementing Alternative 2 is therefore a reduction of (1.6E+2) - (1.7E+1) or 140 person-rem from the existing situation.

Alternative 3: Require plants without an LCS to install a safety-grade LCS with a capacity of ~400 scfh.

This alternative does not alter core damage frequency.

The public risk associated with this option is calculated in much the same manner as used for Alternative 1, as shown in Appendix B. In the existing situation, for internal and non-seismic induced core damage accidents, releases are assumed to be by way of the most passive (and least effective) alternate-treatment method, the isolated steam line. For seismically induced core-damage accidents, the consequences are calculated as though the release is direct to the atmosphere. Twenty of the 21 plants without LCSs are assumed to be characterized by the core damage frequencies used for the BWR 2/3/4 classes while the remaining plant is a BWR 5. The risk per plant-year for the existing situation is 1.8 person-rem for the BWR 2/3/4 and 2.1E-2 for the BWR 5. For the remaining life of the 21 plants the total risk is 9.1E+2 person-rem. The public risk associated with implementing Alternative 3 is 2.9E+2 person-rem. The net effect is (9.1E+2) - (2.9E+2) or a decrease of 6.2E+2 person-rem from the existing situation.

4.1.2 Occupational Exposure

The occupational exposures attributable to the implementation of each of the three alternatives are calculated here. The difference between the existing situation and alternative is expressed as the net effect of implementing the alternative.

Alternative 1: This alternative would result in a one-time increase in occupational exposure incurred during the process of disabling the LCS. Of the 2.5 person-weeks estimated for this effort, half the time (50 person-hours) is estimated to be spent in typical low radiation areas (2.5 mrem/hr). The one-time exposure increase is:

$$(14 \text{ plants})\left(50 \frac{\text{person-hours}}{\text{plant}}\right)\left(2.5\text{E-}3 \frac{\text{rem}}{\text{hr}}\right) = 1.75 \text{ person-rem.} \quad (4.1)$$

The maintenance and surveillance of an LCS is estimated to require 60 person-hours per plant per year. Most of this time is in low-radiation areas for which an average value of 2.5 mrem/hr (ASCE 1986) is used to estimate the total occupational exposure. Assuming 14 plants with the negative pressure LCS and an average remaining life of 34 years (based on an expected lifetime of 40 years from date of first operating license), the averted occupational exposure due to elimination of routine maintenance and surveillance is:

$$(14 \text{ plants})(34 \text{ years})\left(60 \frac{\text{person-hours}}{\text{plant-year}}\right)\left(2.5\text{E-}3 \frac{\text{rem}}{\text{hr}}\right)$$

$$= 71.4 \text{ person-rem.} \quad (4.2)$$

The net occupational exposure impact of implementing Alternative 1 is (71.4) - (1.75) or 69.7 avoided person-rem over 34 years.

Alternative 2: This alternative will result in a one-time increase in occupational exposure during the system upgrade, plus continuing increased occupational exposure due to routine maintenance and surveillance of the upgraded system. Upgrading is estimated to require 6400 person-hours in a low-radiation (2.5 mrem/hr) zone. Surveillance and maintenance of the upgraded system is estimated to require an increase of 1.5 person-weeks/plant-year in the same zone. For 14 plants with an average remaining service life of 34 years, the total occupational exposure impact of implementing Alternative 2 is:

$$\left[(14 \text{ plants})\left(6400 \frac{\text{person-hours}}{\text{plant}}\right)\left(2.5\text{E-}3 \frac{\text{rem}}{\text{hr}}\right) + \left(2.5\text{E-}3 \frac{\text{rem}}{\text{hr}}\right)\right]$$

$$+ \left[(14 \text{ plants})(34 \text{ years})\left(60 \frac{\text{person-hours}}{\text{plant-year}}\right)\left(2.5\text{E-}3 \frac{\text{rem}}{\text{hr}}\right)\right]$$

$$= 2.95\text{E+}2 \text{ person-rem} \quad (4.3)$$

Thus, the impact of implementing Alternative 2 is increased occupational exposure of 295 person-rem.

Alternative 3: This alternative will result in a one-time increase in occupational exposure during installation of the new LCS, plus continuing occupational exposure due to routine maintenance and surveillance of the system.

Installation of the LCS is estimated to require 9600 person-hours. An estimated 60 person-hours per year will be required for maintenance and surveillance, all in a 2.5-mrem/hr zone. For 21 plants and 25 years the total occupational exposure impact of implementing Alternative 3 is:

$$(21 \text{ plants})\left(9600 \frac{\text{person-hours}}{\text{plant}}\right)\left(2.5\text{E-3} \frac{\text{rem}}{\text{hr}}\right)$$

$$+ (21 \text{ plants})(25 \text{ years})\left(60 \frac{\text{person-hours}}{\text{plant-yr}}\right)\left(2.5\text{E-3} \frac{\text{rem}}{\text{hr}}\right)$$

$$= 5.83\text{E+2 person-rem.} \tag{4.4}$$

The impact of implementing Alternative 3 is an increase of 583 person-rem in occupational exposure.

The net occupational exposure impact of implementing each of the alternatives is as follows:

Alternative 1: 6.97E+1 person-rem (avoided exposure)
Alternative 2: 2.95E+2 person-rem (increased exposure)
Alternative 3: 5.83E+2 person-rem (increased exposure)

4.1.3 Industry Implementation Costs

The industry costs associated with implementing each alternative are calculated in this section. These costs are one-time costs incurred by BWR plants to install, upgrade, and/or change their LCSs.

Alternative 1: The costs to remove negative pressure LCSs at the 14 BWRs that have them involve a one-time cost to disable the LCS and to revise technical specifications and operating procedures. The labor to disable an LCS is estimated to be 2.5 person-weeks per plant. To change the documentation is estimated to take 4 person-weeks of engineering/management labor. Using \$40/hr for plant labor and \$50/hr for engineering/managerial labor the cost of implementing alternative 1 is:

$$(14 \text{ plants})[(2.5 \text{ person-weeks/plant})(40 \text{ hr/week})(\$40/\text{hr}) +$$

$$\left(4 \frac{\text{person-weeks}}{\text{plant}}\right)\left(40 \frac{\text{hr}}{\text{week}}\right)(\$50/\text{hr})] = \$168{,}000 \tag{4.5}$$

Alternative 2: There is no documented basis for estimating the cost of a 4000-scfh capacity negative-pressure LCS because one has never been built before. The effort would involve replacing blowers, heaters, valves, and piping from the main steam lines to the standby gas treatment system. The cost of this alternative will instead be based on an NRC estimate of the cost to design and install an LCS that is typical of those now in use (nominal capacity of about 400 scfh). Because this alternative is stated as an upgrade of an existing system, it is assumed that some positions of the existing system could be used in the upgraded system and the cost would be reduced accordingly (assumed to be one-third less than the cost of an entirely new system, as detailed in Alternative 3 below). The costs per plant are summarized below:

Equipment and materials	\$111,000
Labor (160 weeks)(40 hr/week)(\$40/hr)	256,000
Engineering/management (55 weeks)(40 hr/wk)(\$50/hr)	110,000
Total	\$477,000/plant

For the 14 plants with an existing system, the total industry cost would be (14)(\$477,000) = \$6.7 million.

Alternative 3: For those plants without an LCS, this alternative would require the installation of an entirely new LCS with a 400-scfh capacity. The costs cited here are based on the NRC staff's estimate of the installed cost of a 400-scfh system and are 1.5 times those given for Alternative 2.

Equipment and materials	\$166,500
Labor (240 weeks)(40 hr/wk)(\$40/hr)	384,000
Engineering/management (82.5 weeks)(40 hr/wk)(\$50/hr)	165,000
Total	\$715,500/plant

There are 21 BWRs without an LCS at the present time. The total industry implementation cost would thus be: (21)(\$715,500) = \$15.0 million for this alternative.

4.1.4 Industry Operating Costs

In addition to the implementation costs discussed in Section 4.1.3, there will also be changes in industry operations costs. This section describes these operating cost changes from the status quo for each alternative and discounts them to 1988 dollars.

Alternative 1: After disabling the LCS, there would be no operating costs incurred. There would be a savings due to not having to perform routine surveillance and maintenance on the old LCS. An estimated 60 hours per plant per year is required for this function. The total discounted cost for the 14 plants over the 34 years of remaining life is:

$$(14\ \text{plants})(60\ \text{hr/plant-yr})(\$40/\text{hr})(9.609) = \$323{,}000 \qquad (4.6)$$

This is a savings of \$323,000.

Alternative 2: After installation of the 4000-scfh-capacity LCS, routine maintenance and surveillance of the new system will require an increase of 1.5 person-weeks/plant-year. This represents a doubling of the total labor for maintenance and surveillance. For the 14 plants, this will result in a total discounted industry cost of:

$$(14 \text{ plants})(60 \text{ hr/plant-yr})(\$40/\text{hr})(9.609) = \$323{,}000 \qquad (4.7)$$

Alternative 3: The estimated labor increase for routine maintenance and surveillance of the newly installed 400-scfh capacity LCS is 1.5 person-weeks per plant per year. For the 21 plants, this will result in a total discounted industry cost of:

$$(21 \text{ plants})(60 \text{ hr/plant-yr})(\$40/\text{hr})(9.077) = \$457{,}000 \qquad (4.8)$$

4.1.5 NRC Costs

In addition to increased industry costs associated with the three alternatives, the NRC will also incur increased costs. The increased NRC costs for each alternative are calculated in this section.

Alternative 1: The NRC costs to implement this alternative would be the staff time required to formulate and approve the directive to licensees, revise Regulatory Guides, Standard Review Plans and Technical Specifications, and management and legal review. This is estimated at 40 person-weeks. In addition, about 1 additional person-week and \$10,400 per plant would be required for review of changes to plant technical specifications. These cost increases would total the following:

$$\begin{aligned}&(1600 \text{ person-hr})(\$40/\text{person-hr}) + (14 \text{ plants})(\$10{,}400/\text{plant})\\&+ (14 \text{ plants})(40 \text{ person-hr/plant})(\$40/\text{person-hr})\\&= \$232{,}000 \qquad (4.9)\end{aligned}$$

This one-time cost increase would be offset by a cost savings to the NRC due to not having to follow up on operation and maintenance of the existing LCS. The labor saved is estimated at 4 person-hours per plant per year and the total discounted cost saved is (4 person-hr/plant-yr)(\$40/person-hr) (14 plants)(9.609) = \$21,500. Thus, the increased NRC cost for this alternative is \$232,000 - \$21,500 = \$210,500.

Alternative 2: The NRC cost associated with this alternative is estimated to be 12 person-weeks per plant to perform necessary tradeoff studies, develop and justify new requirements, review and approve the requirements, and implement the requirements for MSIV leakage control systems. If any changes to plant technical specifications are required, an NRC cost of \$10,400/plant is estimated, based on an assumption of 4 person-weeks of technical time, 2 person-weeks of management and legal review, and \$800 for Federal Register notices. Assuming technical specification changes will have to be made, the following is the NRC cost for this alternative:

$$(480 \text{ person-hr/plant})(14 \text{ plants})(\$40/\text{person-hr})$$

$$+ (\$10{,}400/\text{plant})(14 \text{ plants}) = \$414{,}400 \quad (4.10)$$

It is estimated that there will be no increase in NRC labor to follow up on operation and maintenance of the larger LCS. Therefore, the total NRC cost increase for this alternative is $414,400.

Alternative 3: The NRC cost to oversee the installation of the 4000-scfh LCSs is estimated to be the same as for Alternative 2: $414,400. In addition to this one-time cost, the NRC will have to follow-up operation and maintenance of the new LCS at each plant. This is estimated to take 3 person-hours per plant per year. To oversee the systems at the 21 plants over 25 years would result in the following discounted cost increase to the NRC:

$$(4 \text{ person-hr})(\$40/\text{person-hr})(21 \text{ plants})(9.077) = \$30{,}500 \quad (4.11)$$

Thus, the total increased cost to the NRC for this alternative is $414,400 + $30,500 or $444,900.

5.0 COST/BENEFIT ANALYSIS SUMMARY

Table 5.1 summarizes the quantified costs and benefits of the three alternatives examined in this report.

TABLE 5.1. Summary of Costs and Benefits

	Alternative 1 (Disable LCS)	Alternative 2 (Upgrade Capacity)	Alternative 3 (Add LCS)
Radiation exposure (person-rem)[a]			
Public	-3.3E+2	1.4E+2	6.2E+2
Occupational	7.0E+1	-2.95E+2	5.83E+2
Total	-2.6E+2	-1.6E+2	3.7E+1
Monetary costs[b]			
Industry implementing costs	$1.68E+5	$6.70E+6	$1.50E+7
Industry operating costs	-$3.23E+5	$3.23E+5	$4.57E+5
NRC Costs	$2.1E+5	$4.14E+5	$4.45E+5
Total	$5.5E+4	$7.4E+6	$1.6E+7

(a) Negative exposure numbers represent increased exposure.
(b) Negative cost numbers represent cost savings.

6.0 REFERENCES

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APPENDIX A

<u>MAIN STEAM ISOLATION VALVE PERFORMANCE DATA</u>
<u>FROM BOILING WATER REACTOR OWNERS' GROUP</u>

APPENDIX A

MAIN STEAM ISOLATION VALVE PERFORMANCE DATA FROM BOILING WATER REACTOR OWNERS' GROUP

The following information was transmitted by letter dated August 15, 1988, to Charles Graves, U.S. NRC, from Donald Grace, Chairman of the Boiling Water Reactor Owner's Group (BWROG). Its subject is the BWROG's MSIV Leakage Closure Committee's recent MSIV leakage history and cost benefit data (revision 2).

BWR OWNERS' GROUP

MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY
AND
COST BENEFIT DATA
(REVISION 2)

COMPILED BY

GE NUCLEAR ENERGY

APRIL 1988

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ROCKWELL - 1

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	HOURS	COMMENTS[4]
1.	5	87	I	0									B,NP
2.	5	87	I	4.5									B,NP
3.	5	87	I	4.1									B,NP
4.	5	87	I	1.9									B,NP
5.	5	87	O	5.3									B,NP
6.	5	87	O	3.1									B,NP
7.	5	87	O	3.5									B,NP
8.	5	87	O	4.1									B,NP
9.	3	87	I	0				185	2.2	80	N		B
10.	4	85	I	0				437	1.6	140	N		B
11.	3	87	O	22.4	0	1		325	0.7	110	N		B
12.	4	85	O	94.0	0	1	22.4	755	0.6	215	N		B
13.	4	87	I	530	0	1		660	3.1	160	N		B
14.	4	85	I	108	0	1	530	437	1.6	140	N		B
15.	3	87	O	48.7	0	1					N		B
16.	4	85	O	0				755	0.6	215	N		B
17.	4	87	I	77.3	0	1		660	3.1	160	N		B
18.	4	85	I	0				437	1.6	140	N		B
19.	3	87	O	0				180	0.3	80	N		B
20.	4	85	O	71.3	0	1	0	1100	1.1	275	N		B
21.	3	87	I	0									B
22.	4	85	I	0				437	1.6	140	N		B
23.	3	87	O	4.7				60	0.1	40	N		B
24.	4	85	O	17.8	0	1	4.7	755	0.6	215	N		B
25.		85	I	0									

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		8
SCFH TOTALS	1000.7	0
BELOW 11.5 SCFH	17	8
BELOW 100 SCFH	23	8

NOTES:

1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ROCKWELL - 2

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	HOURS	COMMENTS[4]
1.		85	O	0									
2.		85	I	0									
3.		85	O	0									
4.		85	I	HIGH	0.7	1							*
5.		85	O	HIGH	0.7	1							*
6.		85	I	0.1									
7.		85	O	0.1									
8.	12	85	I	3.8									NP
9.	11	85	I	0.8									NP
10.	11	85	I	0.5									NP
11.	11	85	I	5.1									NP
12.	12	85	O	3.8									NP
13.	11	85	O	0.8									NP
14.	11	85	O	0.5									NP
15.	11	85	O	5.1									NP
16.	11	86	I	1.9									NP
17.	11	86	I	1.2									NP
18.	11	86	I	9.8									NP
19.	12	86	I	10.1									NP
20.	11	86	O	1.9									NP
21.	11	86	O	1.2									NP
22.	11	86	O	9.8									NP
23.	12	86	O	10.1									NP
24.		85	I	6									
25.		85	I	4									

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		2
SCFH TOTALS	76.6	1.4
BELOW 11.5 SCFH	23	2
BELOW 100 SCFH	23	2

NOTES:
1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ROCKWELL - 3

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	INCREASED[3] OUTAGE LENGTH HOURS	COMMENTS[4]
1.		85	I	21									
2.		85	I	5									
3.		85	O	6									
4.		85	O	4									
5.		85	O	21									
6.		85	O	5									
7.		86	I	2.3									
8.		86	I	2.5									
9.		86	I	5.8									
10.		86	I	2.4									
11.		86	O	2.3									
12.		86	O	2.5									
13.		86	O	5.8									
14.		86	O	2.4									
15.	9	86	I	5									
16.	9	86	I	26	0								
17.	9	86	I	4									
18.	9	86	I	3									
19.	9	86	O	5									
20.	9	86	O	26	0								
21.	9	86	O	4									
22.	9	86	O	3									
23.	10	86	I	3									
24.	10	86	O	3									
25.	10	86	I	6.1									
TOTAL REPAIRS					2								
SCFH TOTALS				176.1	0								
BELOW 11.5 SCFH				21	2								
BELOW 100 SCFH				25	2								

NOTES:

1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenanc practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ROCKWELL - 4

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	INCREASED[3] OUTAGE LENGTH HOURS	COMMENTS[4]
1.	10	86	O	6.1									NP
2.	2	84	I	6.4									NP
3.	2	84	O	6.4									NP
4.	2	84	I	5.3									NP
5.	2	84	O	5.3									NP
6.	2	84	I	9.3									NP
7.	2	84	O	9.3									NP
8.	2	84	I	11.0									NP
9.	2	84	O	11.0									NP
10.	5	85	I	7.7									
11.	5	85	O	7.7									
12.	5	85	I	1.1									
13.	5	85	O	21	1.5		0.7						
14.	5	85	I	1.5									
15.	5	85	O	1.5									
16.	5	85	I	8.4									
17.	5	85	O	8.4									
18.	5	86	I	26.4	0.9		1.4						
19.	5	86	O	0.9									
20.	5	86	I	0.7									
21.	5	86	O	0.7									
22.	5	86	I	10.5									
23.	5	86	O	10.5									
24.	5	86	I	30	0.3		6.9						
25.	5	86	O	0.3									

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		3
SCFH TOTALS	207.4	2.7
BELOW 11.5 SCFH	22	3
BELOW 100 SCFH	25	3

NOTES:

1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

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BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ROCKWELL - 5

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	INCREASED[3] OUTAGE LENGTH HOURS	COMMENTS
1.	5	87	I	1.4									
2.	5	87	O	1.4									
3.	5	87	I	27	10.5								
4.	5	87	O	10.5									
5.	5	87	I	85	1.3								
6.	5	87	O	1.3									
7.	5	87	I	6.9									
8.	5	85	O	6.9									
9.	8	85	I	0.0									
10.	8	85	O	0.0									
11.	8	85	I	0.6									
12.	8	85	O	0.6									
13.	8	85	I	0.5									
14.	8	85	O	0.5									
15.	8	85	O	44.9	3.4		3.8						
16.	9	85	I	3.4									
17.	9	85	I	9.5									
18.	9	85	O	9.5									
19.	2	87	I	11.5	0.0								
20.	2	87	O	HIGH	0.0								
21.	5	87	I	0.0									
22.	5	87	O	0.0									
23.	2	87	I	0.0									
24.	2	87	O	42.4	0.0	2							
25.	2	87	I	17.2	0.0	0		0					FLUSH ON

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		7
SCFH TOTALS	281	15.2
BELOW 11.5 SCFH	18	7
BELOW 100 SCFH	24	7

NOTES:
1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenanc practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ROCKWELL - 6

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	HOURS	COMMENTS[4]
1.	2	87	O	0.0									
2.	2	87	I	3.8									
3.	2	87	O	3.8									
4.	12	85	I	2.9									
5.	12	85	O	2.9									
6.	12	85	I	2.4									
7.	12	85	O	2.4									
8.	12	85	I	0.0									
9.	12	85	O	0.0									
10.	12	85	I	6.4									
11.	12	85	O	6.4									
12.	3	86	I	0.0									
13.	3	86	O	0.0									
14.	3	86	I	HIGH	5.9	0		0					FLUSH ONLY
15.	3	86	O	5.9									
16.	3	86	I	19.4	0.0	0		0					FLUSH ONLY
17.	3	86	O	0.0									
18.	3	86	I	6.5									
19.	3	86	O	53.7	6.5	0		0					FLUSH ONLY
20.													
21.													
22.													
23.													
24.													
25.													

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		3
SCFH TOTALS	1165	12.4
BELOW 11.5 SCFH	16	3
BELOW 100 SCFH	18	3

NOTES:
1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ATWOOD-MORRILL - 1

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	HOURS	COMMENTS[4]
1.	4	86	I	0.7									N
2.	4	86	I	2.1									N
3.	4	86	I	1.3									N
4.	4	86	I	2.7									N
5.	4	86	O	7.5									N
6.	4	86	O	11.5	0.4	1		225	1.0	112	N		N
7.	4	86	O	3.7									N
8.	4	86	O	10.4									N
9.	11	87	I	>43	0.9	1		120		48			NP,*
10.	11	87	O	>37	0.9	1		120					NP,*
11.	11	87	I	>43	6.4	1		180		60			NP,*
12.	11	87	O	>43	6.4	1		120		48			NP,*
13.	11	87	O	>43	1.6	1		120		48			NP,*
14.	11	87	I	>43	1.6	1		120		48			NP,*
15.	10	87	I	26									NP
16.	12	84	I	2.2									NP
17.	3	85	I	5.6									
18.	3	87	I	0.1									
19.	6	87	I	4.1									
20.	3	86	O	7.3									
21.	11	87	I	1.7									
22.	10	87	I	0.9									
23.	9	87	O	2.6									
24.	10	87	O	42.8	1.6	1							N
25.	9	86	I	3.1									N

TOTAL REPAIRS		8
SCFH TOTALS	159.1	19.8
BELOW 11.5 SCFH	15	8
BELOW 100 SCFH	19	8

NOTES:
1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ATWOOD-MORRILL - 2

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	INCREASED[3] OUTAGE LENGTH HOURS	COMMENTS[4]
1.	10	85	I	15.2									N
2.	9	86	I	0.5									N
3.	3	85	I	0.1									N
4.	9	86	I	3.7									N
5.	11	85	I	1.6									N
6.	9	86	I	3.8									N
7.	11	85	I	1.3									N
8.	9	86	O	3.8									N
9.	10	85	O	0									N
10.	9	86	O	0.5									N
11.	3	85	O	0.1									N
12.	9	86	O	3.7									N
13.	11	85	O	2.7									N
14.	9	86	O	HIGH	0	1		222	10		N		N,*
15.	11	85	O	HIGH	0.9	1	HIGH	1100	40		Y		N,*
16.	2	84	I	2.2									B
17.	3	85	I	2.1									B
18.	3	87	I	.02									B
19.	6	87	I	6.4									B
20.	2	84	I	6.0									B
21.	4	85	I	HIGH	0.5	1	HIGH	94	0.6		N		B,*
22.	7	87	I	HIGH	4.1	5		288	4.6		N		B,*
23.	2	84	I	0.1									B
24.	2	85	I	3.9									B
25.	3	87	I	.02									B

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		4
SCFH TOTALS	57.7	5.5
BELOW 11.5 SCFH	20	4
BELOW 100 SCFH	21	4

NOTES:

1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ATWOOD-MORRILL - 3

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	HOURS	COMMENTS[4]
1.	5	87	I	4.8									B
2.	2	84	I	6.0									B
3.	4	85	I	4.1									B
4.	3	87	I	.02									B
5.	6	87	I	3.5									B
6.	2	84	O	14.7	2.3	1	2.1	82	0.3		N		B
7.	10	87	I	28.4	0.1	1		200	0.03	40	N		B,NP
8.	10	87	I	>30	4.3	1		300	0.04	60	N		B,*,NP
9.	10	87	I	4.8									B,NP
10.	10	87	I	>30	1.2	1		300	0.04	60	N		B,*,NP
11.	10	87	O	>30	0.2	1		200	0.03	40	N		B,*,NP
12.	10	87	O	5.2									B,NP
13.	10	87	O	4.0									B,NP
14.	10	87	O	>30	3.6	1		200	0.03	40	Y	48	B,*,NP
15.	3	85	O	2.1									B
16.	3	87	O	7.1									B
17.	6	87	O	6.4									B
18.	2	84	O	10.7									B
19.	4	85	O	HIGH	0.5	1	1.1	92	0.2		N		B*
20.	6	87	O	1.1									B
21.	7	87	O	4.1									B
22.	2	84	O	1.6									B
23.	2	85	O	3.9									B
24.	3	87	O	4.3									B
25.	5	87	O	4.8									B

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		7
SCFH TOTALS	121.6	12.2
BELOW 11.5 SCFH	18	7
BELOW 100 SCFH	20	7

NOTES:
1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ATWOOD-MORRILL - 4

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	INCREASED[3] OUTAGE LENGTH HOURS	COMMENTS[4]
1.	2	84	O	2.4									B
2.	3	85	O	HIGH	4.1	1	3.0	97	0.2		N		B,*
3.	3	87	O	3.0									B
4.	6	87	O	3.5									B
5.	2	85	I	82.3	1.8	1	108.6	213	2.3		N		B
6.	7	85	I	108.6	0.6	2	5.3	163	1.5		N		B
7.	4	87	I	5.3									B
8.	2	85	I	4.0									B
9.	7	85	I	1.7									B
10.	4	87	I	0.6									B
11.	2	85	I	1.4									B
12.	7	85	I	4.3									B
13.	4	87	I	5.1									B
14.	2	85	I	HIGH	0.1	1	HIGH	290	3.1		N		B,*
15.	7	85	I	HIGH	0.7	1	HIGH	144	1.7		N		B,*
16.	4	87	I	HIGH	9.4	2		291	2.9		N		B,*
17.	2	85	O	16.6	0.6	1	7.8	150	0.6		N		B
18.	7	85	O	7.8									B
19.	4	87	O	1.1									B
20.	2	85	O	0.1									B
21.	7	85	O	5.9									B
22.	4	87	O	5.3									B
23.	2	85	O	5.7									B
24.	7	85	O	2.4									B
25.	4	87	O	2.3									B

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		7
SCFH TOTALS	269.4	17.3
BELOW 11.5 SCFH	18	7
BELOW 100 SCFH	20	7

NOTES:

1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER ATWOOD-MORRILL - 5

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	HOURS	COMMENTS[4]
1.	2	85	O	1.6									B
2.	2	85	O	13.6	7.1	1	9.5	156	0.4		N		B
3.	7	85	O	9.5	0.7	1	5.8	87	0.3		N		B
4.	4	87	O	5.8									B
5.	5	87	O	2.7									B
6.	5	87	O	1.1									B
7.	5	87	I	35	4.8	1		734	0.04		N		N,NP
8.	5	86	I	1.8									N,NP
9.	5	86	I	>42.3	0.3	1	>42.3	1225	0.07		N		N,NP,*
10.	5	87	I	>42.3	3.0	2		1113	0.06		N		N,NP,*
11.	5	86	I	>42.3	1.6	1	37.6	1134	0.04		N		N,NP,*
12.	5	87	I	37.6	7.8	1		694	0.05		N		N,NP
13.	5	87	I	>42.3	3.6	1		754	0.06		N		N,NP,*
14.	5	86	O	1.8									N,NP
15.	5	87	O	1.6									N,NP
16.	5	86	O	0.3									N,NP
17.	5	87	O	2.8									N,NP
18.	5	86	O	1.1									N,NP
19.	5	87	O	3.1									N,NP
20.	5	86	O	>42.3	8.0	1	>42.3	1240	0.03		N		N,NP,*
21.	5	87	O	>42.3	3.6	2		1282	0.08		N		N,NP,*
22.													
23.													
24.													
25.													

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		10
SCFH TOTALS	119.4	40.5
BELOW 11.5 SCFH	12	10
BELOW 100 SCFH	15	10

NOTES:
1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER CRANE - 1

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR* SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	HOURS	COMMENTS
1.	4	83	I	1.3									
2.	3	85	I	4.1									
3.	4	83	O	1.3									
4.	3	85	O	4.1									
5.	4	83	I	42.8	4.3	1	1.3	2		2			
6.	3	85	I	1.3									
7.	4	83	O	42.8	4.3	1	1.3	2		2			
8.	3	85	O	1.3									
9.	4	83	I	2.6									
10.	3	85	I	2.8									
11.	4	83	O	2.6									
12.	3	85	O	2.8									
13.	4	83	I	1.3									
14.	3	85	I	2.7									
15.	4	83	O	1.3									
16.	3	85	O	2.7									
17.	4	84	I	2.6									
18.	7	86	I	2.7									
19.	4	84	O	2.6									
20.	7	86	O	2.7									
21.	4	84	I	0									
22.	7	86	I	2.7									
23.	4	84	O	0									
24.	7	86	O	2.7									
25.	4	84	I	0									

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		2
SCFH TOTALS	133.8	8.6
BELOW 11.5 SCFH	23	2
BELOW 100 SCFH	25	2

NOTES:
1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER CRANE - 2

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	INCREASED[3] OUTAGE LENGTH HOURS	COMMENTS[4]
1.	7	86	I	1.3									
2.	4	84	O	0									
3.	7	86	O	1.3									
4.	4	84	I	0									
5.	7	86	I	1.3									
6.	4	84	O	0									
7.	7	86	O	1.3									
8.	1	86	I	0									
9.	1	86	O	146	1.2		16.3						
10.	1	86	I	0									
11.	1	86	O	18.4	0		2.3						
12.	1	86	I	0									
13.	1	86	O	48.4	0		4.6						
14.	1	86	I	5.8									
15.	1	86	O	5.8									
16.	9	87	I	16.3	10.4								
17.	9	87	O	16.3	10.4								
18.	9	87	I	2.3									
19.	9	87	O	2.3									
20.	9	87	I	4.6									
21.	9	87	O	4.6									
22.	9	87	I	3.5									
23.	9	87	O	3.5									
24.	3	85	I	2.3									
25.	3	85	O	2.3									

	INITIAL SCFH	AFTER REPAIR SCFH
TOTAL REPAIRS		5
SCFH TOTALS	287.6	22
BELOW 11.5 SCFH	20	5
BELOW 100 SCFH	24	5

NOTES:
1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenanc practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY AND COST BENEFIT DATA

MSIV MANUFACTURER CRANE - 3

	MONTH	YEAR	INBOARD[1] OR OUTBOARD	INITIAL SCFH	AFTER REPAIR SCFH	NUMBER OF REWORKS	NEXT[2] OUTAGE INITIAL SCFH	REPAIR MH	REMS	TOTAL HRS REPAIR TIME	INCREASED[3] OUTAGE LENGTH Y/N	INCREASED[3] OUTAGE LENGTH HOURS	COMMENTS[4]
1.	3	85	I	0									
2.	3	85	O	13.8	4.6		4.6						
3.	3	85	I	6.9									
4.	3	85	O	6.9									
5.	3	85	I	0									
6.	3	85	O	158	9.2		37.8						
7.	10	86	I	2.3									
8.	10	86	O	2.3									
9.	10	86	I	4.6									
10.	10	86	O	4.6									
11.	10	86	I	9.2									
12.	10	86	O	9.2									
13.	10	86	I	3.5									
14.	10	86	O	37.8	0								
15.													
16.													
17.													
18.													
19.													
20.													
21.													
22.													
23.													
24.													
25.													

TOTAL REPAIRS		3
SCFH TOTALS	259.1	13.8
BELOW 11.5 SCFH	11	3
BELOW 100 SCFH	13	3

NOTES:
1. I - Inboard, O = Outboard
2. If repair is required
3. Y = Yes, N = No, U = Unknown
4. B = BWROG modifications incorporated, N = BWROG modifications not incorporated, NP = new plant data
5. * = Leak rate not known; however, valve was returned to acceptable leak rates with normal maintenance practices. (Data not included in initial leakage average.)

BWROG MSIV LEAKAGE CLOSURE COMMITTEE

RECENT MSIV LEAKAGE HISTORY SUMMARY

MSIV Manufacturer	Rockwell	Atwood-Morrill	Crane	Totals	
NUMBER OF PLANTS	11	9	4	24	
TOTAL INITIAL TESTS	144	121	64	329	
TOTAL INITIAL TESTS MINUS TESTS WITH >X SCFH REPORTED	140	96	64	300	
TOTAL INITIAL SCFH	1858.3	727.2	814.3	3399.8	
AVERAGE INITIAL SCFH/TEST	13.3	7.6	12.7	11.3	
TOTAL INITIAL TESTS > ALLOWED LEAKAGE	25	36	10	71	
TOTAL AFTER REPAIR SCFH	31.7	95.3	44.4	171.4	
AVERAGE AFTER REPAIR SCFH/TEST	1.3	2.6	4.4	2.4	
OF INITIAL TESTS:					
TOTAL <11.5 SCFH	117	83	54	254	(77%)
TOTAL <100 SCFH	138	95	62	295	(≥90%)

APPENDIX B

PUBLIC RISK CALCULATIONS

APPENDIX B

PUBLIC RISK CALCULATIONS

The public risk associated with the existing situation can be calculated using current estimates of core-damage frequency (per plant-year), the number of plants with and without installed LCSs, the current estimates of MSIV performance, and the dose consequences of MSIV leakage at various rates.

For the BWR 4 and older classes (abbreviated BWR 2/3/4), the core-damage frequency from internal and nonseismically induced external events is 2.0E-4 per plant-year (NRC 1982); for seismically-induced core-damage events, the probability is 8.0E-5 per plant-year (Hatch 1987). The core-damage frequency probabilities for BWR 5 and 6 plants (abbreviated BWR 5/6) are 3.7E-5 (NRC 1981) and 6.0E-7 (NRC 1989) per plant-year for internal and seismically induced core-damage events.

The typical LCS is useful only as a consequence mitigation method if the total leakage from all steam lines is less than about 400 scfh. Those leakage treatment (consequence mitigation) strategies that make use of nonseismic category I equipment are not considered likely to be available if the core damage is caused by a seismic event. Thus, the total public risk from MSIV leakage at a plant equipped with an LCS is the sum of incremental risks due to the different combinations core damage cause, MSIV leak rate and equipment availability. The contributors to this total are shown in Figure B.1.

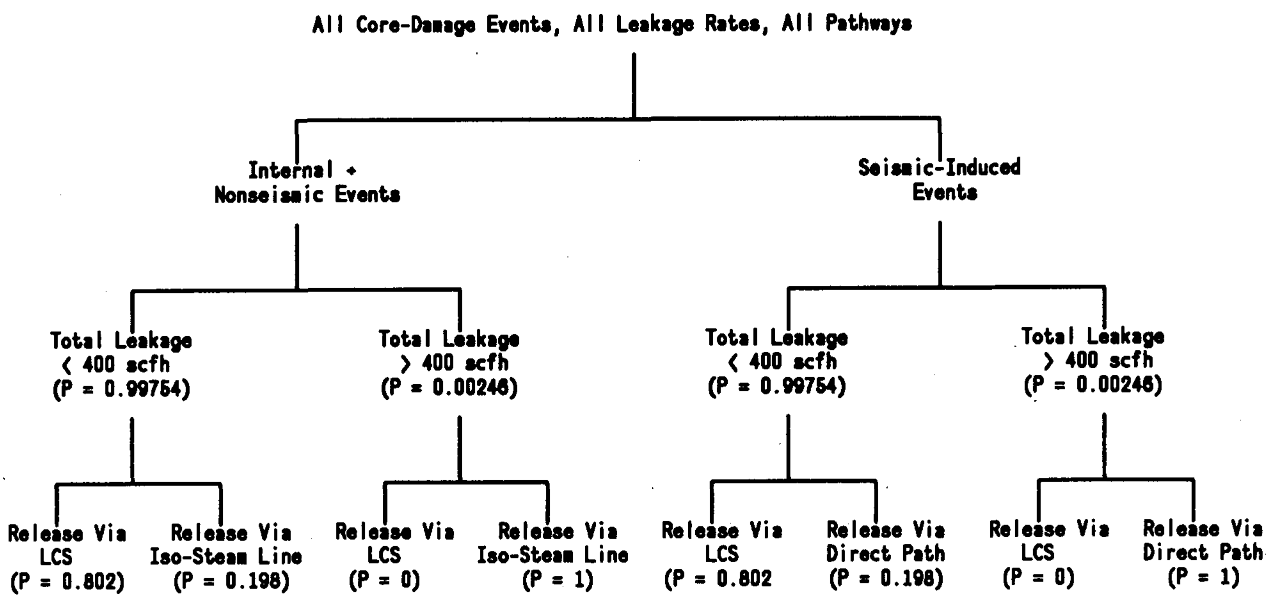

FIGURE B.1. Contributions to Total Public Risk from MSIV Leakage

The total public risk from all core-damage events is the sum of risk due to the internal and nonseismic event-induced core-damage events (abbreviated "I+NS") and that due to seismic event-induced core-damage events (abbreviated "SE"). For the I+NS core-damage events, the LCS is assumed to be used to mitigate the release if the total leakage is less than about 400 scfh (P = 0.99754) and if the LCS is available (P = 0.802). If the LCS is not available (P = 1 - 0.802, or 0.198), the "isolated steam line" passive alternate treatment method described in Appendix C is assumed to be used.

If total MSIV leakage exceeds the 400-scfh capacity of the LCS (P = 0.00246) the availability of the LCS becomes zero and the release is assumed to be via the isolated steam line path with an availability of 1.

For the SE core-damage events, the LCS is assumed to be used to mitigate the release if the total leakage is less than about 400 scfh and if the LCS is available. If the LCS is not available, the leakage is assumed to be direct to atmosphere without benefit of filtration or holdup in the steam lines. The presumption is that the steam lines, not being seismic Category I, would fail in any earthquake severe enough to produce core damage. If total MSIV leakage the exceeds 400-scfh capacity of the LCS, the availability of the LCS becomes zero and the release is assumed to be direct to the atmosphere.

The plants without an LCS are assumed to use the isolated steam line release path for the I+NS core-damage accidents at all leakage rates. For the core-damage accidents induced by seismic events, the steam lines are assumed to fail and the release is direct to the atmosphere.

The public risk for each of these increments of core-damage probability, equipment availability and leakage rate can be determined and summed to get the total public risk. To determine the probability of total MSIV leakage falling into a given range, a Monte Carlo simulation of four main steam lines, each with two valves in series, was conducted using the Boiling Water Reactor Owners' Group (BWROG) leak test data base to define the performance of individual valves. (The BWROG data are presented in Appendix A.) The simulation method and results are detailed in Appendix C. The simulation established that the probability of total leakage from four steam lines, each with two valves in series, being less than 46 scfh (an average of 11.5 scfh per line) is 0.91117. The probability of the total being between 46 and 400 scfh is 0.08637, and the probability of total leakage exceeding 400 scfh is 0.00246. The public risk for each of these leakage rates can be bounded by multiplying the probability that total leakage will fall within a given range times the population dose that results from leakage at the upper end of the range via the specified pathway. The population doses for each leak rate and pathway are given in Table B.1. These values were calculated using CRAC2 (Ritchie 1983) and the method described in Section 4 of NUREG-1169 (NRC 1986).

This integration of public risk contributions from all core damage causes, leak rates and pathways is the sum of the risk from plants with LCSs and from those without LCSs. For the plants that have an LCS, the risk can be expressed as the sum in Equation (B.1). The public risk from plants without an LCS can be expressed as shown in Equation (B.2).

$$
\begin{aligned}
\text{Total Risk} = {} & \left\{ \begin{pmatrix}\text{Internal +}\\ \text{nonseismic-induced}\\ \text{core-damage frequency}\end{pmatrix} \left[\begin{pmatrix}\text{Prob. of total}\\ \text{leakage}\\ < 46\text{ scfh}\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at}\\ 46\text{ scfh via LCS}\end{pmatrix} + \begin{pmatrix}\text{Prob. of total}\\ \text{leakage}\\ > 46\text{ but} < 400\end{pmatrix} \begin{pmatrix}\text{Consequence}\\ \text{of 400 scfh}\\ \text{via LCS}\end{pmatrix} \right] \begin{pmatrix}\text{Availability}\\ \text{of LCS}\end{pmatrix} \right\} \\
& + \left\{ \begin{pmatrix}\text{Internal +}\\ \text{nonseismic-induced}\\ \text{core-damage frequency}\end{pmatrix} \left[\begin{pmatrix}\text{Prob. of total}\\ \text{leakage}\\ < 46\text{ scfh}\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 46 scfh}\\ \text{via iso-steam line}\end{pmatrix} + \begin{pmatrix}\text{Prob. of total}\\ \text{leakage}\\ > 46\text{ but} < 400\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 400 scfh}\\ \text{via iso-steam line}\end{pmatrix} \right] \begin{pmatrix}1 - \text{Availability}\\ \text{of LCS}\end{pmatrix} \right\} \\
& + \left\{ \begin{pmatrix}\text{Internal +}\\ \text{nonseismic-induced}\\ \text{core-damage frequency}\end{pmatrix} \left[\begin{pmatrix}\text{Prob. of total}\\ \text{leakage} > 400\\ \text{but} < 4000\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 4000 via}\\ \text{iso-steam line}\end{pmatrix} \right] \begin{pmatrix}1\end{pmatrix} \right\} \\
& + \left\{ \begin{pmatrix}\text{Seismic-induced}\\ \text{core-damage frequency}\end{pmatrix} \left[\begin{pmatrix}\text{Prob. of total}\\ \text{leakage}\\ < 46\text{ scfh}\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 46}\\ \text{via LCS}\end{pmatrix} + \begin{pmatrix}\text{Prob. of total}\\ \text{leakage} > 46\\ \text{but} < 400\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 400}\\ \text{via LCS}\end{pmatrix} \right] \begin{pmatrix}\text{Availability}\\ \text{of LCS}\end{pmatrix} \right\} \\
& + \left\{ \begin{pmatrix}\text{Seismic-induced}\\ \text{core-damage frequency}\end{pmatrix} \left[\begin{pmatrix}\text{Prob. of total}\\ \text{leakage}\\ < 46\text{ scfh}\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 46}\\ \text{via direct path}\end{pmatrix} + \begin{pmatrix}\text{Prob. of total}\\ \text{leakage} > 46\\ \text{but} < 400\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 400}\\ \text{via direct path}\end{pmatrix} \right] \begin{pmatrix}1 - \text{Availability}\\ \text{of LCS}\end{pmatrix} \right\} \\
& + \left\{ \begin{pmatrix}\text{Seismic-induced}\\ \text{core-damage frequency}\end{pmatrix} \left[\begin{pmatrix}\text{Prob. of total}\\ \text{leakage} > 400\\ \text{but} < 4000\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 4000}\\ \text{via direct path}\end{pmatrix} \right] \begin{pmatrix}1\end{pmatrix} \right\} \qquad \text{(B.1)}
\end{aligned}
$$

$$\text{Total Risk} = \left\{ \begin{pmatrix}\text{Internal +}\\ \text{nonseismic-induced}\\ \text{core-damage frequency}\end{pmatrix} \left[\begin{pmatrix}\text{Prob. of total}\\ \text{leakage}\\ \text{46 scfh}\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 46 scfh}\\ \text{via iso-steam line}\end{pmatrix} \right. \right.$$

$$+ \begin{pmatrix}\text{Prob. of total}\\ \text{leakage} > 46\\ \text{but} < 400\end{pmatrix} \begin{pmatrix}\text{Consequence of leakage}\\ \text{at 400 scfh via}\\ \text{iso-steam line}\end{pmatrix}$$

$$\left. \left. + \begin{pmatrix}\text{Prob. of total}\\ \text{leakage} > 400\\ \text{but} < 4000\end{pmatrix} \begin{pmatrix}\text{Consequence of leakage}\\ \text{at 4000 scfh via}\\ \text{iso-steam line}\end{pmatrix} \right] \right\}$$

$$+ \left\{ \begin{pmatrix}\text{Seismic-induced}\\ \text{core-damage}\\ \text{frequency}\end{pmatrix} \left[\begin{pmatrix}\text{Prob. of total}\\ \text{leakage}\\ \text{46 scfh}\end{pmatrix} \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 46}\\ \text{via direct path}\end{pmatrix} \right. \right.$$

$$\begin{pmatrix}\text{Prob. of total}\\ \text{leakage} > 46\\ \text{but} < 400\end{pmatrix} \begin{pmatrix}\text{Consequence of leakage}\\ \text{at 400 scfh via}\\ \text{direct path}\end{pmatrix}$$

$$\left. \left. + \begin{pmatrix}\text{Prob. of total}\\ \text{leakage} > 400\\ \text{but} < 4000\end{pmatrix} \begin{pmatrix}\text{Consequence of leakage}\\ \text{at 4000 scfh via}\\ \text{direct path)}\end{pmatrix} \right] \right\} \tag{B.2}$$

Thus, using the method and factors introduced here, the public risk (per plant-year) associated with a single BWR 2/3/4 with an LCS is expressed in Equations (B.3) and (B.4):

$$\begin{aligned}
&\{(2.0E\text{-}4)[(0.91117)(1.1E{+}2 \text{ person-rem}) +\\
&(0.08637)(9.7E{+}2 \text{ person-rem})](0.802)\} +\\
&\{(2.0E\text{-}4)[(0.91117)(3.5E{+}1 \text{ person-rem}) +\\
&(0.08637)(1.3E{+}3 \text{ person-rem})](0.198)\} +\\
&\{(2.0E\text{-}4)[(0.00246)(2.8E{+}4 \text{ person-rem})](1)\} +\\
&\{(8.0E\text{-}5)[(0.91117)(1.1E{+}2 \text{ person-rem}) +\\
&(0.08637)(9.7E{+}2 \text{ person-rem})](0.802)\} +\\
&\{(8.0E\text{-}5)[(0.91117)(1.2E{+}4 \text{ person-rem}) +\\
&(0.08637)(1.04E{+}5 \text{ person-rem})](0.198)\} +\\
&\{(8.0E\text{-}5)[(0.00246)(1.04E{+}6)](1)\},
\end{aligned} \tag{B.3}$$

or

$$\begin{aligned}
&(3.0E\text{-}2) + (5.7E\text{-}3) + (1.4E\text{-}2) + (1.2E\text{-}2) +\\
&(3.2E\text{-}1) + (2.0E\text{-}1) = 5.9E\text{-}1 \text{ person-rem/plant-year}
\end{aligned} \tag{B.4}$$

TABLE B.1. Population Doses by Various MSIV Leakage Paths

Leakage Pathway	Total Leak Rate (4 lines)	Person-rem Per Event
Leakage control system	46	1.1E+2
	400	9.7E+2
	4000(a)	9.7E+3(a)
	4000(a)	2.8E+4(b)
Isolated Steam line	46	3.5E+1
	400	1.3E+3
	4000	2.8E+4
Direct Release	46	1.2E+4
	400	1.04E+5
	4000	1.04E+6

(a) Assuming that an upgraded LCS has capacity to treat 4000 scfh, population dose is scaled up by 10x over that for 400 scfh.
(b) Existing LCSs would be unable to handle 4000 scfh. Thus, the population dose is the same as by the isolated steam line pathway.

TABLE B.2. Summary of Key Frequency and Risk Values

	BWR 2/3/4		BWR 5/6	
Internal and nonseismic core-damage frequency (per plant year)	2.0E-4		3.7E-5	
Seismic event-induced core-damage frequency (per plant year)	8.0E-5		6.0E-7	
	W/LCS	W/O LCS	W/LCS	W/O LCS
Population risk per reactor rear (person-rem)	5.9E-1	1.8E0	1.3E-2	2.1E-2

The public risk per plant year associated with a single BWR 2/3/4 without an LCS is:

$$\{(2.0E-4)[(0.91117)(3.5E+1 \text{ person-rem}) +$$
$$(0.08637)(1.2E+3) + (0.00246)(2.8E+4 \text{ person-rem})]\} +$$
$$\{(8.0E-5)[0.91117)(1.2E+4 \text{ person-rem}) +$$
$$(0.08637)(1.04E+5 \text{ person-rem}) +$$
$$(0.00246)(1.04E+6 \text{ person-rem})]\} \quad (B.5)$$

or, $$(4.3E-2) + (1.8E0) = 1.8E0 \text{ person-rem/plant year.} \quad (B.6)$$

For BWR 5/6 plants, the core-damage frequency due to internal and nonseismic events is 3.7E-5/plant-year, with the seismic-induced core-damage frequency being 6.0E-7/plant-year. Substituting these values for the corresponding BWR 2/3/4 values above, the public risk per plant year for a BWR 5/6 with an LCS is 1.3E-2 person-rem/plant-year. Similarly, the public risk for a BWR 5/6 without an LCS (of which there is only one) is 2.1E-2 person-rem/plant-year. After calculating the average remaining life for plants with and without LCSS, based on an expected life of 40 years from issuance of first operating license, the total public risk for all plants with LCSs is as shown in Table B.3.

TABLE B.3. Public Risk for BWRs With and Without Leakage Control Systems

BWR 2/3/4 with LCS	
(8 plants)(34 years)(5.9E-1 person-rem/plant year) =	1.6E+2 person-rem
BWR 5/6 with LCS	
(6 plants)(34 years)(1.3E-2 person-rem/plant year) =	2.7 person-rem
Total Risk, plants with LCS	1.6E+2 person-rem
BWR 2/3/4 without LCS	
(20 plants)(25 years)(1.8E0 person-rem/plant year) =	9.1E+2 person-rem
BWR 5/6 without LCS	
(1 plant)(38 years)(2.1E-2 person-rem/plant year) =	8.0E-1 person-rem
Total Risk, plants without LCS	9.1E+2 person-rem
Total Risk, All Plants	1.1E+3 person-rem

Alternative 1. Require plants with negative pressure LCSs to take them out of service. The public risk for this alternative is calculated by assuming that for the 14 affected plants, all MSIV leakage is released by the least effective alternate pathway (isolated steamline) for the accident cases when the steam lines would be assumed intact (internal and nonseismic external event-induced core-damage events).

For seismically induced core-damage events, any release is assumed to be direct to the atmosphere without benefit of holdup in the steamlines. The total risk can be represented in Equation (B.7):

$$
\begin{aligned}
\text{Total Risk} = {} & \left(\begin{array}{l}\text{Internal +}\\ \text{Nonseismic-induced}\\ \text{core-damage frequency}\end{array}\right)\left[\left(\begin{array}{l}\text{Prob. of}\\ \text{total leakage}\\ < 46 \text{ scfh}\end{array}\right)\left(\begin{array}{l}\text{Consequence of}\\ \text{leakage at 46 scfh}\\ \text{via iso. steam line}\end{array}\right)\right.\\
& + \left(\begin{array}{l}\text{Prob. of total}\\ \text{leakage} > 46\\ \text{but} < 4000 \text{ scfh}\end{array}\right)\left(\begin{array}{l}\text{Consequence of}\\ \text{leakage at 400 scfh}\\ \text{via iso. steam line}\end{array}\right) + \left(\begin{array}{l}\text{Prob. of}\\ \text{leakage} > 400\\ \text{but} < 4000\end{array}\right)\\
& \left.\left(\begin{array}{l}\text{Consequence of leak-}\\ \text{age at 4000 scfh}\\ \text{via iso. steam line}\end{array}\right)\right] + \left(\begin{array}{l}\text{Seismic-induced}\\ \text{core-damage}\\ \text{frequency}\end{array}\right)\left[\left(\begin{array}{l}\text{Prob. of}\\ \text{total leakage}\\ < 46 \text{ scfh}\end{array}\right)\right.\\
& \left(\begin{array}{l}\text{Consequence of}\\ \text{leakage at 46}\\ \text{via direct path}\end{array}\right) + \left(\begin{array}{l}\text{Prob. of}\\ \text{leakage} > 46\\ \text{but} < 400 \text{ scfh}\end{array}\right)\left(\begin{array}{l}\text{Consequence of}\\ \text{leakage at 400 scfh}\\ \text{via direct path}\end{array}\right)\\
& \left. + \left(\begin{array}{l}\text{Prob. of}\\ \text{leakage} > 400\\ \text{but} < 4000 \text{ scfh}\end{array}\right)\left(\begin{array}{l}\text{Consequence of leakage}\\ \text{at 4000 scfh via}\\ \text{direct path}\end{array}\right)\right] \qquad \text{(B.7)}
\end{aligned}
$$

For a single BWR 2/3/4, the risk per plant year is shown in Equation (B.8):

$$
\begin{aligned}
& \{(2.0E\text{-}4)[(0.91117)(3.5E\text{+}1 \text{ person-rem}) + (0.08637)\\
& (1.3E\text{+}3 \text{ person-rem})(0.00246)(2.8E\text{+}4 \text{ person-rem})]\}\\
& + \{(8.0E\text{-}5)[(0.91117)(1.2E\text{+}4 \text{ person-rem}) + (0.08637)\\
& (1.04E\text{+}5 \text{ person-rem}) + (0.00246)(1.04E\text{+}6 \text{ person-rem})]\}\\
& = 1.8 \text{ person-rem/plant year} \qquad \text{(B.8)}
\end{aligned}
$$

For a single BWR 5/6, the corresponding risk would be 2.2E-2 person-rem/plant year.

Total risk for 8 BWR 2/3/4 plants and 6 BWR 5/6 plants is shown in Equations (B.9) and (B.10):

$$\text{BWR 2/3/4:}\quad (8\ \text{plants})(34\ \text{years})\left(1.8\ \frac{\text{person-rem}}{\text{plant-year}}\right) = 4.9\text{E+2 person-rem} \tag{B.9}$$

$$\begin{array}{ll} \text{BWR 5/6:} & (6\ \text{plants})(34\ \text{years})\left(2.2\text{E-2}\ \frac{\text{person-rem}}{\text{plant-year}}\right) = \underline{4.5\text{E+0 person-rem}} \\ \text{Total} & \qquad\qquad\qquad\qquad\qquad\qquad\qquad\quad 4.9\text{E+2 person-rem} \end{array} \tag{B.10}$$

Therefore, the net effect of this alternative is to increase the risk from the 14 plants equipped with negative-pressure LCSs from 1.6E+2 to 4.9E+2 person-rem, an increase of 330 person-rem.

Alternative 2. Require plants with negative pressure LCS to upgrade them to higher capacity (~4000 scfh). The public risk for this alternative is calculated by assuming that all MSIV leakage is released by the LCS pathway; i.e., there would be no direct releases via the ISO steam line pathway.

As documented in Table C.4 in Appendix C, this alternative would provide greater than 99.75% assurance that seismic Category I LCSs have sufficient capacity to handle the combined leakage from all MSIVs.

The risk per plant year is shown in Equation (B.11):

$$\begin{aligned} \text{Risk/plant-year} = & \begin{pmatrix}\text{Total core-}\\ \text{damage frequency}\end{pmatrix} \Bigg[\begin{pmatrix}\text{Probability of leakage}\\ < 46\ \text{scfh}\end{pmatrix} \\ & \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 46 scfh}\\ \text{via LCS}\end{pmatrix} + \begin{pmatrix}\text{Probability of leakage}\\ > 46\ \text{but} < 400\ \text{scfh}\end{pmatrix} \\ & \begin{pmatrix}\text{Consequence of}\\ \text{leakage at 400 scfh}\\ \text{via LCS}\end{pmatrix} + \begin{pmatrix}\text{Probability of leakage}\\ > 400\ \text{but} < 4000\ \text{scfh}\end{pmatrix} \\ & \begin{pmatrix}\text{Consequence of Leakage}\\ \text{at 4000 scfh via LCS}\end{pmatrix} \Bigg] \end{aligned} \tag{B.11}$$

For a single BWR 2/3/4 plant, the risk is shown in Equation (B.12):

$$\begin{aligned} & \{(2.8\ \text{E-4})[(0.91117)(1.1\text{E+2 person-rem}) + \\ & (0.08637)(9.7\text{E+2 person-rem}) + \\ & (0.00246)(9.7\text{E+3 person-rem})]\} = 5.8\text{E-2}\ \frac{\text{person-rem}}{\text{plant-year}} \end{aligned} \tag{B.12}$$

For a single BWR 5/6 with total core-damage frequency of 3.8E-5/plant year, the public risk is 7.9E-3 person-rem/plant-year. Total risk for 8 BWR 2/3/4 plants and 6 BWR 5/6 plants is shwon in Equations (B.13) and (B.14):

$$\text{BWR 2/3/4:}\quad (8 \text{ plants})(34 \text{ years})(5.8\text{E-}2\ \frac{\text{person-rem}}{\text{plant-year}}) = 1.6\text{E+}1 \text{ person-rem} \tag{B.13}$$

$$\text{BWR 5/6:}\quad (6 \text{ plants})(34 \text{ years})(7.9\text{E-}3\ \frac{\text{person-rem}}{\text{plant-year}} = \underline{1.6\text{E0 person-rem}}$$
$$\text{Total}\qquad 1.7\text{E+}1 \text{ person-rem} \tag{B.14}$$

Therefore, the net effect of this Alternative is to reduce the risk from the 14 plants with LCSs from 1.6E+2 to 1.7E+1 person-rem, a decrease of 140.

<u>Alternative 3</u>. Require plants without an LCS to install a safety-grade LCS with capacity comparable to those now in service. For the existing situation, the public risk per plant-year is the same as was calculated for Alternative 1. For internal and nonseismic core-damage events, all releases are assumed to be by way of the most passive (and least effective) alternate treatment method, the isolated steam line. For seismically induced core-damage events, the consequences are calculated as though the release is direct to the atmosphere. Twenty of the 21 plants currently without LCSs are assumed to be characterized by the core-damage frequencies used previously for the BWR 2/3/4 classes. The risk per plant year for the existing situation is 1.8 person-rem. For 20 plants without LCSs and 25 years average remaining life, the total risk is:

$$(20 \text{ plants})(25 \text{ years})(1.8\ \frac{\text{person-rem}}{\text{plant-year}}) = 9.0\text{E+}2 \text{ person-rem} \tag{B.15}$$

The twenty-first plant without an LCS is a BWR 5 with an estimated 38 years remaining life. The risk is:

$$(1 \text{ plant})(38 \text{ years})(2.2\text{E-}2\ \frac{\text{person-rem}}{\text{plant-year}}) = 8.4\text{E-}1 \text{ person-rem} \tag{B.16}$$

Total risk for plants without LCSs is

$$(9.0\text{E+}2) + (8.4\text{E-}1) = 9.0\text{E+}2 \text{ person-rem} \tag{B.17}$$

The public risk associated with requiring plants that do not have an LCS to install one comparable to those now in service (nominal capacity of 400 scfh) is calculated in the same manner as that used for the current condition in Alternative 1.

For BWR 2/3/4 plants with an LCS the risk per plant year is 5.9E-1 person-rem. For the 20 plants that would be required to add an LCS and an average of 25 years remaining life, the risk is:

$$(20 \text{ plants})(25 \text{ years})(5.9\text{E-}1 \frac{\text{person-rem}}{\text{plant-year}}) = 2.9\text{E+}2 \text{ person-rem} \quad (B.18)$$

For one BWR 5 plant, the risk is:

$$(1 \text{ plant})(38 \text{ years})(1.3\text{E-}2 \frac{\text{person-rem}}{\text{plant-year}}) = 4.9\text{E-}1 \text{ person-rem}$$

The total risk is

$$(2.9\text{E+}1) + (4.9\text{E-}1) = 2.9\text{E+}2 \text{ person-rem.} \quad (B.19)$$

The net effect of implementing Alternative 3 would be to reduce risk from the 21 plants without LCSs from 9.1E+2 to 2.9E+2 person-rem, a decrease of 620 person-rem.

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TABLE B.4. Operating BWRs and LCS Status

No LCS	BWR Class	Negative Pressure LCS	BWR Class
Oyster Creek	2	Duane Arnold	4
Nine Mile Point 1	2	Hatch 2	4
Nine Mile Point 2	5	Fitzpatrick	4
Dresden 2	3	Susquehanna 1	4
Dresden 3	3	Susquehanna 2	4
Millstone 1	3	Shoreham	4
Monticello	3	Limerick 1	4
Quad Cities 1	3	Limerick 2	4
Quad Cities 2	3	WNP 2	5
Pilgrim	3	LaSalle 1	5
Browns Ferry 1	4	LaSalle 2	5
Browns Ferry 2	4	Grand Gulf	6
Browns Ferry 3	4	Perry 1	6
Vermont Yankee	4	Clinton	6
Peach Bottom 2	4	Total 14 BWRs	
Peach Bottom 3	4		
Cooper	4	**Positive Pressure LCS**	**BWR Class**
Hatch 1	4	Fermi 2	4
Brunswick 1	4	Hope Creek	4
Brunswick 2	4	River Bend	6
Big Rock Point	NA	Total 3 BWRs	
Total 21 BWRs			

APPENDIX C

UPDATED PROBABILISTIC RISK ASSESSMENT

APPENDIX C

UPDATED PROBABILISTIC RISK ASSESSMENT

This appendix presents a summary and update of the probabilistic risk assessment first presented in NUREG-1169 (NRC 1986) to estimate the public risk due to MSIV leakage and the offsite dose associated with using each of the leakage control pathways.

A review of the MSIV leakage information provided by the BWR Owners' Group (BWROG) shows that performance continues to improve. A summary of valve performance data is given in Table C.1.

TABLE C.1. MSIV Performance Summary

Leak Rate, scfh	Number of Valves Tested		
	Inboard	Outboard	Both Inboard and Outboard
< 11.5	127	128	255
11.5 to 19.9	5	8	13
20 to 29.9	6	4	10
30 to 39.9	3	1	4
40 to 49.9	1	6	7
50 to 59.9	0	1	1
60 to 69.9	0	0	0
70 to 79.9	1	1	2
80 to 89.9	2	0	2
90 to 99.9	0	1	1
100 to 109.9	2	0	2
110 to 499.9	0	2	2
500 to 550	1	0	1
> 550	0	0	0
	148	152	300
>30	2	2	4
>37	0	1	1
>42.3	4	2	6
>43	3	2	5
"High"	7	6	13

Although the amount and apparent quality of the BWROG data far exceed any that have been heretofore available, there are some concerns about the data:

- The same leakage value often appears to be assigned to both inboard and outboard valves in series. This is apparently done when the measured leak rate through both valves (from the pressurized inter-space) is less than the leak rate specification. This practice could overstate the actual leak rate of a given valve, but it will conservatively represent the sealing capability of two valves in series in a steam line.

- There appear to be several older plants represented in the data with leakage problems that may distort the industry average leakage distribution upwards.

- There appear to be several new plants with initial leakage problems, perhaps due to manufacturing or installation problems. This could again distort the distribution towards higher values than would be observed in a group of mature plants.

- A number of test results are "greater than" some leakage rate or simply "high," apparently due to limitations of the testing equipment. These readings play a vital role in defining the probability of leakage in excess of 11.5 scfh, and thus must be factored into any proposed distribution in some fashion. Without discrete data, however, these points cannot be fit to traditional distributions (i.e., log-normal, gamma, etc.). They can, however, be used in a simple frequency block distribution.

Considering the above facts and limitations, it was decided that the data for inboard and outboard valves will be combined into one population because no physical reason for different reliability performance is known. The potential for several "bad" plants distorting the data towards a higher leakage rate will be noted, but all the available plant data will be used. For the 29 data points reported as "greater than" some value, it will be assumed that the data follow the same distribution as those valves for which specific leak rate values greater than 30 scfh were reported. No attempt was made to fit this data to a known frequency distribution (i.e., log-normal or gamma distribution). Rather, a simple histogram model was used within the leakage ranges given in Tables C.2 and C.3.

TABLE C.2. Overall Leakage Probability Distribution from BWROG Data

Leakage, scfh	All MSIVs	Probability	Cumulative Probability
L < 11.5	255/329	0.7751	0.7751
11.5 ≤ L < 20	13/329	0.0395	0.8146
20 ≤ L < 30	10/329	0.0304	0.8450
L > 30	51/329	0.1550	1.0000

TABLE C.3. Leakage Probability Distribution for the 22 Leak Rates Greater Than 30 scfh

Leakage, scfh	All MSIVs	Probability	Cumulative Probability
30 ≤ L < 40	(4/22)(0.1550)	0.0282	0.8732
40 ≤ L < 50	(7/22)(0.1550)	0.0493	0.9225
50 ≤ L < 100	(6/22)(0.1550)	0.0423	0.9648
100 ≤ L < 1000	(5/22)(0.1550)	0.0352	1.0000

TOTAL LEAKAGE INTO FOUR MAIN STEAM LINES

Using the intervals in Table C.2, the total leakage into four main steam lines was simulated, assuming two valves in series per line. A simple random number model was used to associate probability and leak rate, as summarized in Table C.4. Two random numbers are chosen independently, to represent the inboard and outboard MSIV leakage. The leakage rate was assigned as in Table C.4. Because the valves are in series, the smaller of the two leakage terms was taken as leakage into one steam line. This process was repeated four times to represent four steam lines, again assuming full independence of the four lines. The leakage into the four lines was then summed, representing the total leakage rate for that particular closure event of the MSIVs and recorded. The process was repeated many times (n = 100,000), generating a frequency distribution for total leakage into four steam lines. The results are given in Table C.5.

The capacity of the typical leakage control system is about equal to 100 scfh per steam line, or 400 scfh total. Thus, the LCS plays a role for leakage greater than 11.5 scfh, but less than about 400 scfh.

TABLE C.4. Assignment of Leakage Rate Based on Random Number Occurrence

Random Number	Leakage Interval Assumed, scfh	Leakage Rate Assigned, scfh
0 to 0.7751	0 to 11.5	linear from 0 to 11.5
0.7751 to 0.8146	11.5 to 20	linear from 11.5 to 20
0.8146 to 0.8450	20 to 30	linear from 20 to 30
0.8450 to 0.8732	30 to 40	linear from 30 to 40
0.8732 to 0.9225	40 to 50	linear from 40 to 50
0.9225 to 0.9648	50 to 100	linear from 50 to 100
0.9648 to 1.000	100 to 1000	linear from 100 to 1000

TABLE C.5. Leakage Distribution for Steam into Four Main Steam Lines

Leakage Range, scfh	Probability	Cumulative Probability
0 to 11.5	0.11943	0.11943
11.5 to 20	0.39334	0.51277
20 to 30	0.30839	0.82116
30 to 40	0.06923	0.89039
40 to 50	0.03464	0.92503
50 to 100	0.06447	0.98950
100 to 200	0.00607	0.99557
200 to 300	0.00110	0.99667
300 to 400	0.00087	0.99754
> 400	0.00246	1.0000

The BWR 6 design includes an additional main steam line shutoff valve downstream of the MSIVs. The leakage performance of these slow-closing valves is thought to be at the very least equal to that of an MSIV. This would effectively lower the probability of steam line leakage in a BWR 6 at least one order of magnitude below the estimate used here for two MSIVs per steam line. The presence of such valves will be ignored in this analysis to determine the need for any leakage control measure downstream of two MSIVs per steam line.

IMPLICATIONS OF CONTAINMENT FAILURE FOR IMPORTANCE OF MSIV LEAKAGE

Leakage of MSIVs could play an important role if the containment remains intact, and if there is a significant source term available for release under such conditions. The source term available for release depends on the accident scenario. Some sequences result in a direct blowdown to the suppression pool with subsequent entrainment of fission products in the pool. Other scenarios result in substantial aerosol generation in the drywell atmosphere, which is available for release. Any calculation of the importance of the MSIV leakage must consider the following factors:

- recognition that source terms will vary with core-melt scenario due to plate-out and entrainment of fission products in the containment and suppression pool
- the existing probability of containment failures that would compete with or exceed any MSIV leakage
- the potential that MSIV leakage might reduce the probability of containment failure given core-melt
- the likelihood that core-melt will result in containment failure that would make risk from MSIV leakage insignificant by comparison.

Source Terms Varying with Core-Melt Scenario

The source term present in the steam lines that is available for release through the MSIVs will be a function of the core-melt scenario. The recirculation line break that bypasses a direct path to the suppression pool would likely have a higher fraction of the fission product inventory remain in the drywell region compared to scenarios that allow a direct flow path to the suppression pool. This effect is characterized in Table C.6, which gives the estimated distribution of CsI fission products for several representative core-melt scenarios in a BWR Mark III containment.

TABLE C.6. Distribution of CsI in the BWR Mark III Containment for Various Core-Melt Scenarios

Accident Sequence[a]	Fraction of CsI Core Inventory by Location				
	RCS	Drywell	Pool	Containment	Environment
TC	0.19	3.6E-02	0.77	1.9E-04	6.8E-03
TPI	8.4E-02	3.9E-03	0.91	7.5E-07	2.4E-04
TQUV	6.3E-02	3.8E-06	0.94	6.8E-04	8.4E-04
SE	9.1E-02	1.2E-02	0.89	2.0E-03	3.3E-03

(a) Accident sequence designation from WASH-1400, Reactor Safety Study (NRC 1975).

In all cases, most of the CsI inventory released from the core ends up in the suppression pool. The inventory in the RCS that may leak past the MSIVs can vary by up to a factor of 3 compared to a TC scenario (transient with failure to SCRAM). The frequency of this scenario is very low, however, and it is a relatively insignificant contributor to overall plant risk. The fraction of CsI in the RCS for the other scenarios is quite similar.

Containment Failures Competing with MSIV Leakage

This scenario introduces the potential that other containment failures may already be in existence at the time of the core-melt scenario and would compete with the MSIV leakage. A review of past experience in LWRs indicates that the probability for leakage on the order of several percent of containment volume per day is approximately 0.01, or roughly the same probability as having two MSIVs in the same steam line leaking at greater than 45 scfh. Thus, frequency of a core-melt with pre-existing containment leakage is thought to be approximately the same as for a core-melt with MSIV leakage. However, with these events being independent, it could be assumed that only 1% of core-melts with MSIV leakage also would have containment leakage from an existing failure. As a result, existing failures may be comparable to, but would not reduce appreciably, the risk of MSIV leakage.

MSIV Leakage Reducing Risk of Core-melt Containment Failure

This scenario brings up the possibility that containment leakage, such as through the MSIVs, can act as a pressure release and possibly reduce the probability of massive containment failure in some scenarios. A review of containment failures, however, indicates that gross containment failure is currently assumed for leakage rates approaching 100% volume per day. This represents penetration failures with effective hole diameters up to approximately 6 in. With leakage past the MSIVs from 11.5 to 500 scfh, or approximately 2E+02 to 1E+04 scfd, this represents from 1% to 10% of the containment volume per day for plants with containment volumes of 1E+05 and 1E+06 ft^3, respectively. Furthermore, leakage through such failures is currently not thought to affect the dynamic behavior of systems in any appreciable way, such as action of the suppression pool vents and delivery of water to sumps during the accident. As a result, at this time it is not thought that MSIV leakage at the leakage rates of interest plays any role in avoiding massive containment failure.

Potential of Containment Failure by Core-melt

Current PRAs assume that there is a probability that containment will suffer a gross failure as a result of core-melt. This would make any contribution from MSIV leakage negligible to overall risk.

The survival of containment from gross ruptures or bypasses and, just as importantly, from smaller failures thus becomes important to this issue. The probabilistic analysis of potential public risk and benefit from plant modifications has historically been approached conservatively. For most safety issues, this approach has translated into a pessimistic prediction of the ability of containment to survive the core-melt accident intact. Early

PRAs (NRC 1975) assigned a very high probability to containment failure due to steam or hydrogen explosion and overpressure. Later studies acknowledged the likely better performance of containment and shifted the emphasis about failure probabilities towards less drastic ruptures (i.e., containment leakage). However, for many accident sequences, they still assigned a very high probability to containment failure by some means.

The potential also exists for significant variation in failure probability between the BWR Mark I, II, and III containments. The designs differ greatly in suppression pool configuration and volume, in impact of internal venting characteristics on accident dynamics, and in volume and construction techniques used for the primary and secondary containments. Both gross containment failure (or bypass) modes and smaller leakage rates are of interest, because, in evaluating the potential risk contribution to the public following a core damage accident, any other release pathway from containment could very easily overshadow any contribution from MSIV leakage. A report on testing of containment for leakage (Zapp 1968) indicated that a 1/16-in.-diameter hole would result in 0.1% containment volume leakage per day from a 1E+06 ft^3 volume at accident conditions (i.e., 55 psi and 150°F), or approximately 40 scfh. This is on the same order of magnitude as the design leakage capacity of the LCS.

This issue does not in any fashion reduce the potential frequency of core-melt scenarios or reduce the severity of the accidents once they have proceeded to core-melt. It does, however, deal directly with the issue of containment leakage. As such, a gross overestimate of containment failure probabilities would underestimate the potential importance of this issue.

It is recognized that the performance of containment will play a significant role in determining whether leakage past the MSIVs presents a notable safety concern. However, such a fundamental plant safety feature as containment should not simply be assumed to fail, thus putting this issue in the trivial category. Therefore, for purposes of this analysis of MSIV leakage consequences and the effectiveness of the leakage control system, it was assumed that containment remains intact.

FREQUENCY OF CORE-MELT EVENTS WHERE MSIV LEAKAGE IS OF INTEREST

Three classes of internal (nonseismic) accidents leading to core damage without containment failure were used to evaluate the importance of MSIV leakage: the recirculation line breaks, steam line breaks, and transients. The frequencies associated with these events are presented in Table C.7.

As presented in Table C.7, the total predicted core damage frequency due to internal and non-seismic events is estimated to be 3.70E-05 per plant-year for the BWR 6 plant. This is assumed to be representative of plants with the newer Mark II and Mark III containments. The estimate for the BWR 4 plant with Mark I containment is 2.00E-04 per plant-year.

TABLE C.7. Summary of BWR Core Damage Frequencies[a]

BWR 6	
Recirculation line break	2.45E-06
Steam line break	2.45E-06
Transients	3.20E-05
Total	3.70E-05
BWR 4	
Recirculation line break	6.70E-07
Steam line break	5.70E-07
Transients	2.00E-04
Total	2.00E-04

(a) Assuming base case core-melt frequency of one per plant year.

The major contributors to the frequency estimates in Table C.7 are transient-initiated core damage scenarios that typically involve eventual loss of coolant injection or decay heat removal functions, which have no bearing on the question of the need for an LCS. Consideration of the recirculation line break alone would reduce the above estimates of core damage frequency where MSIV leakage may be of importance by an order of magnitude.

AVAILABILITY OF MSIV LEAKAGE CONTROL PATHWAYS

This section presents a discussion of the availability of MSIV leakage control pathways. For the purposes of this study, the MSIVs are assumed to leak following a core damage accident.

Following a LOCA and reactor trip, the MSIVs are expected to close. The same conditions apply to the turbine bypass valve (TBV) and the turbine control valve (TCV). The configuration of valves for the reference plant, the BWR 5 with Mark II containment, was determined from the plant Final Safety Analysis Report (WPPSS 1984) and discussions with licensee's operator training personnel. Steam line drain isolation valves must also be considered. These air-operated valves will fail to open on loss of offsite power (LOOP) and eventual loss of compressed air supply. The system consists of four TBVs, two TCVs and one steam line drain isolation valve (IV) that must be properly aligned for the pathway of interest. The "closed" state of the MSIVs, TBV, and TCV indicates that, theoretically, no leakage through the steam system should occur. Experience has shown, however, that the MSIVs do leak and this leakage might exceed the typical standard 11.5 scfh leakage limit.

In the following subsections, a detailed discussion of each of the leakage control pathways is provided. An availability assessment of these pathways is also presented by the means of fault tree analysis. The probability of successful operation (availability) of the leakage control pathways, frequency of initiating event, and total public risk estimates are also provided. An attempt as also been made to point out the benefits and the limitations associated with each one of the leakage control pathways.

Role of Loss of Offsite Power During a Design Basis Accident

Of particular concern to this program is the potential availability of leakage control pathways during the design basis accident (DBA). This again consists of a recirculation line break leading to core damage. Such conditions in the early stages of the accident in the plant will also lead to turbine trip and LOOP. Several of the proposed pathways are, however, highly dependent on this power source. As a result, the performance of the various leakage control pathways under loss of offsite power conditions becomes important.

The general approach in establishing regulatory requirements for control of radionuclide releases is to ensure that the proposed system will perform its intended function under all reasonably probable accident conditions. This approach then sets performance standards under presumed conditions, such as loss of offsite power. A probabilistic evaluation of the issue, however, must take into consideration the probability of such additional failures in support systems. Thus, the potential for a plant shutdown initiating loss of offsite power will be reviewed here along with its impact on the leakage control pathways.

Loss of Offsite Power

A recent review of loss of offsite power events by the Electric Power Research Institute (EPRI) indicates that through 1983 there were 47 reported LOOP events at 52 plant sites (Wyckoff 1984). With 533 site-years observed, this represents an average of 0.088 events per site-year. This number falls to 0.027 events per site year for the years 1981 through 1983 due to specific site improvements in electrical switchyard reliability. The median duration for the loss of power was approximately 1/2 hour, with the longest being approximately 9 hours. For outages exceeding 1/2 hour, the average was 0.049 events per site-year. However, this represents an industry average, with actual incidents being very site-specific. Of the 52 plant sites studied, one-half (26 of 52) had not experienced a loss of offsite power. Fifteen sites have had only one event, indicating that the remaining 11 sites experienced 32 losses of offsite power. As a result, several of the sites discussed in the report have undertaken equipment upgrades to improve power reliability, which is reflected in improved performance in recent years.

For this study, the failure mode of interest is the potential for plant shutdown following a LOCA, which would initiate grid instabilities leading to loss of offsite power. There have been several recorded instances of such events, most notably at Turkey Point, which has two 666-MWe PWR units. Turkey Point has experienced six events, with grid instability problems and interactions between the two units contributing to the outages. A major switchyard upgrade is expected to eliminate interactions between the units.

The experience at Turkey Point would represent an upper bound on the potential for inducing loss of offsite power following plant trip.

The transient shutdown data for plants operating at between 26% and 110% power provides a history of shutdown frequency of 6.01/plant-year for PWRs and 6.16/plant-year for BWRs. The two Turkey Point units went into operation in December 1972 and September 1973, giving a total operational time of 11 and 10.25 plant-years, to talling 21.25 plant-years up to the end of 1983. Thus, the total number of trips during this period, using the average PWR trip frequency at power, is

$$(6.01 \text{ trips/yr})(21.5 \text{ yr}) = 129.22 \text{ trips} \tag{C.1}$$

It will be further assumed that all of the six recorded events of loss of offsite power at the site were due to plant shutdowns. The conditional probability of loss of offsite power given a plant shutdown at Turkey Point can then be estimated as

$$\frac{6}{129.22} = 0.05 \tag{C.2}$$

The potential for plant shutdown leading to loss of offsite power is thus highly site-dependent, ranging historically from 0 to 0.05, with the probability put at 0.05/shutdown at the site which has displayed the highest tendency to cause grid instabilities on plant shutdown. However, the true industry average is somewhere between these extremes.

The value of 0.05 for loss of offsite power on plant shutdown will be used here as a measure for the potential states of plant equipment after a LOCA-induced shutdown.

Power Recovery. Several of the leakage control pathways under consideration here will use equipment downstream of the MSIVs that will be powered by non-safety-grade electric buses. As such, this equipment will be vulnerable to LOOP events with the plant shutdown and no longer maintaining normal house loads. The potential for recovery of offsite power during the course of the accident then becomes of interest. Given that the median time for recovery of offsite power has been approximately 1/2 hour, recovery would mean that the systems in question would be available within 1/2 hour from the time of trip with a probability of approximately 0.5. Progression of accidents to core damage in less than 1/2 hour following a LOOP event will essentially make the systems available for leakage control measures. The net effective probability of loss of systems due to loss of offsite power is

$$(0.05)(0.5) = 0.025 \tag{C.3}$$

This calculation is used in the following development of availabilities of the leakage control options.

Impact of Loss of Offsite Power on Leakage Control Options. In the leakage control options to follow, some of the valves and equipment to be used will not be available given a LOOP, including compressed air supplies of the balance of plant which may affect air-powered valve positions, vacuum pumps, steam supplies, etc. To account for the potential functional loss of some valves and equipment during a LOOP event, the systems-loss probability estimated above (0.025) will be included in any estimate of systems' lack of availability due to mechanical failure.

Leakage Control System Pathway

The main steam line isolation valve leakage control system (MSIV-LCS) is designed to minimize the amount of fission products that could escape to the environment. For a negative pressure system, this is accomplished by directing the leakage past the closed MSIVs to a bleedline using a blower that directs the leakage into the reactor building and eventually through the standby gas treatment system. Thus, leakage through the MSIV is processed by the SGTS prior to release to the atmosphere. Use of this pathway provides some treatment and holdup of the radioactive gases by the SGTS charcoal beds and HEPA filters. It also provides for an elevated release from the plant.

One drawback associated with this pathway is the limited capacity of the system, which isolates at flows exceeding about 100 scfh per main steam line. Therefore, if a large MSIV leakage occurs, this pathway is not available. The conditional probabilities of individual MSIV leakage that were analyzed earlier in this appendix lead to a prediction that the total leakage past the valves for four steam lines in parallel will exceed 400 scfh only 0.25% of the time. The LCS is a safety-related system, and its equipment will be unaffected by a LOOP.

The potential release flow path for operation of a negative pressure MSIV-LCS is presented in Figure C.1. The leakage passes through the inboard and outboard MSIV-LCS, the SGTS system, and the stack before it is finally released to the atmosphere.

A success tree illustrates the successful operation of this leakage control pathway (Figure C.2). The top event of this tree is a core damage event with successful operation of the LCS. For this to occur, three main conditions must be met:

- closure of the MSIVs with leakage within the LCS capacity
- operator action to initiate LCS operation
- successful mechanical operation of the LCS components, including proper initial alignment of valves.

The first condition that must be fulfilled is that the MSIVs must leak within the capacity of the LCS. If an average of 100 scfh per MSL is exceeded, the LCS will isolate and this pathway will not be available. For purposes of this pathway availability analysis, a probability of 1 is assigned to this condition.

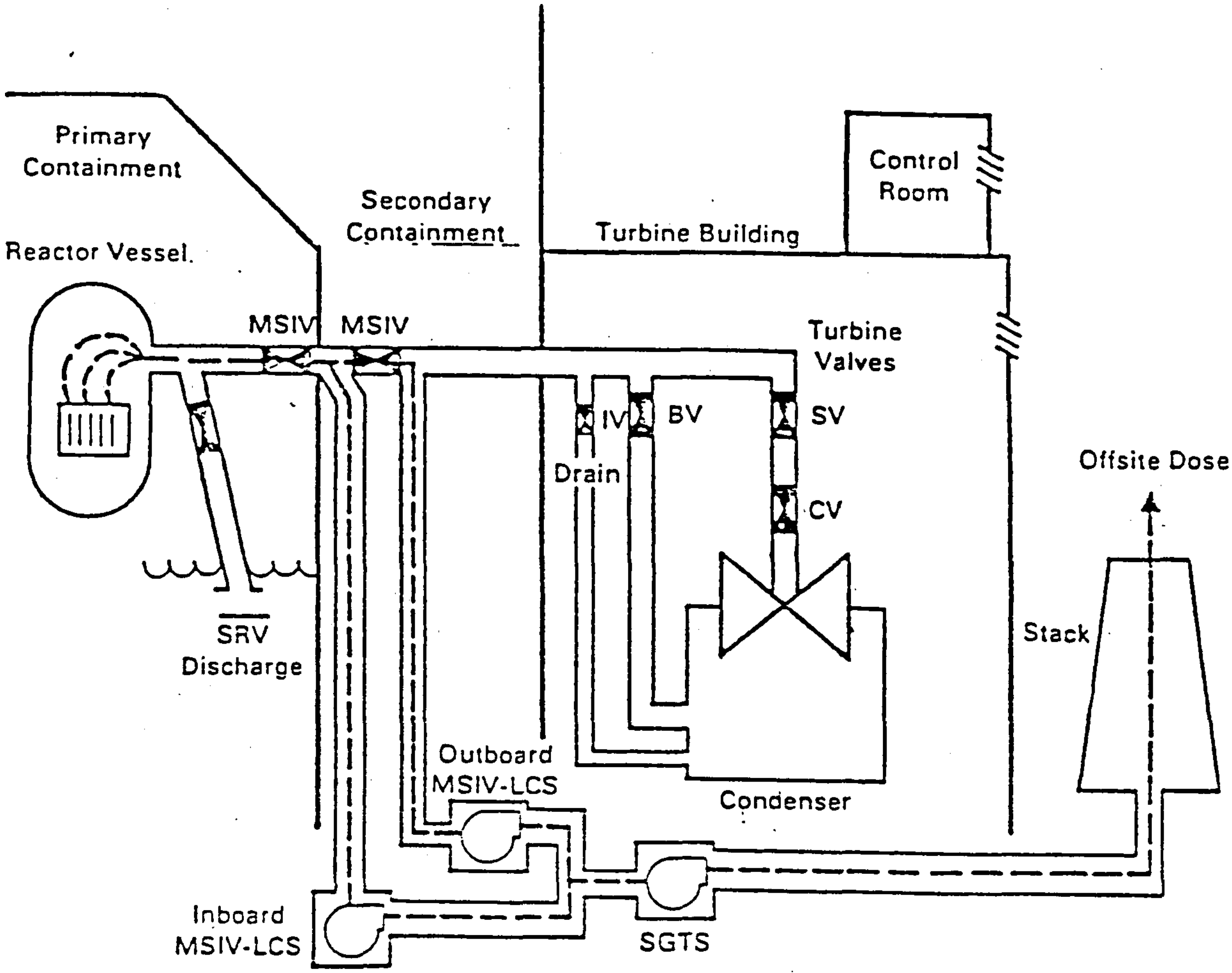

FIGURE C.1. LCS Flow Path

The second condition in this analysis that needs to be met is the operator interaction. (Not all of the pathways identified require some operator action.) During the scenario progression to core damage, operator action to initiate LCS operation is unlikely. Discussions with the reference plant operator trainers indicate that responding to the system demands to prevent core damage will require most of the operating staff's time and attention. However, the existing procedures on detection of radiation in the steam lines would dictate use of the LCS. The operator will attempt to control MSIV leakage with a high probability at some time after core damage, but given the continued confusion and demands on operators, it is uncertain if the LCS would be started immediately after leakage and the presence of radiation in the steam lines is noted. Since the identified MSIV leakage control pathway will be documented in the emergency procedures, a probability of 0.9 is assigned that the operator will take some action, and that he will do it correctly.

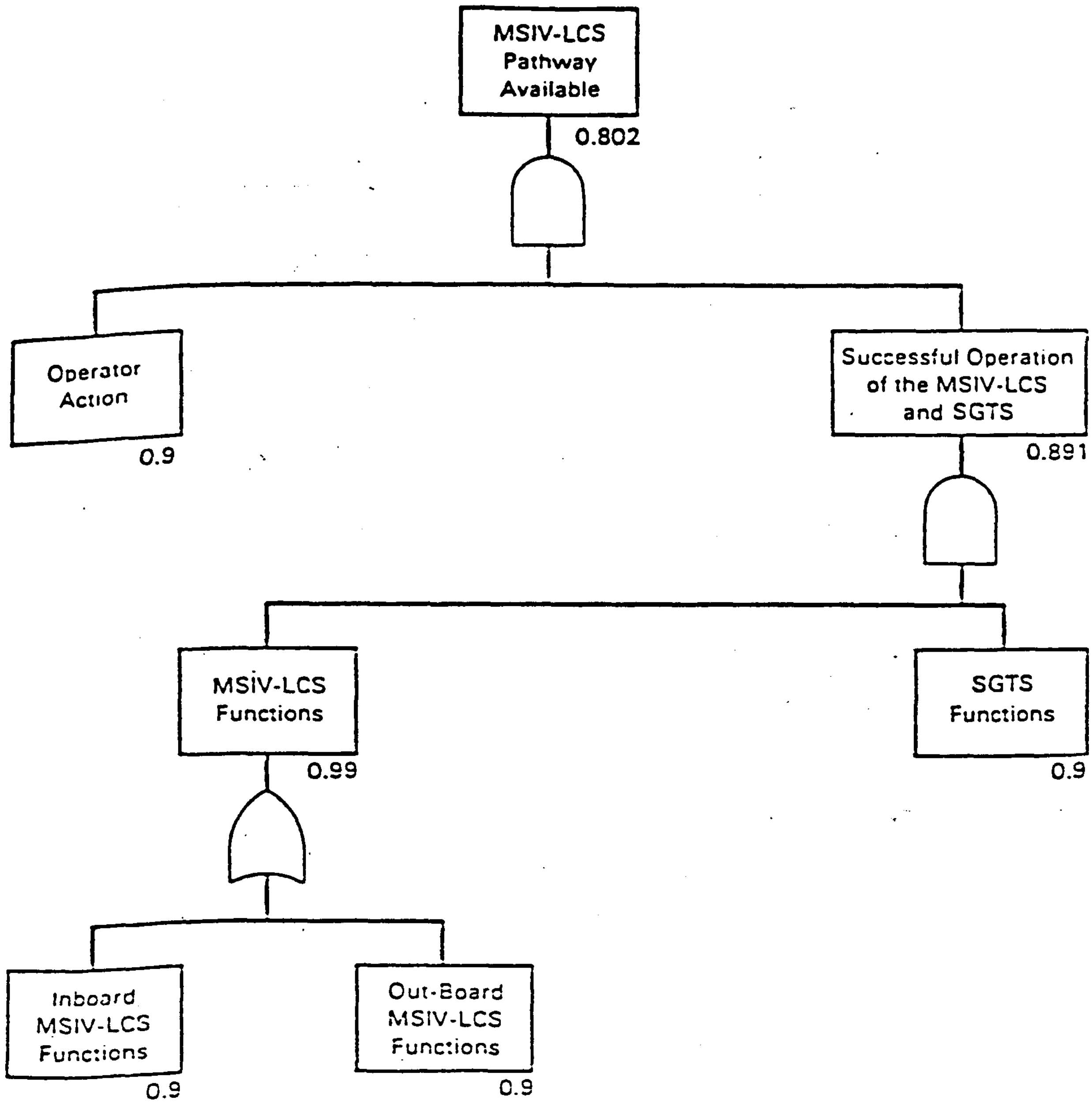

FIGURE C.2. LCS Availability Success Tree

The third condition that must be met for successful operation of the LCS is successful operation of the MSIV-LCS and the SGTS. The LCS consists of valves, a mechanical blower, power supply, and switching components. In the absence of any specific availability data, a value of 0.9 is assumed for availability on demand of the blower for the inboard and outboard LCS. If operation of only one LCS is required for success, each having a reliability of 0.9, the overall probability of success for the inboard or outboard LCS is

$$(0.9) + (0.9) - (0.9)(0.9) = 0.99 \tag{C.4}$$

The same basic value of 0.9 is also used for the SGTS system. Therefore, the total probability of availability of the MSIV-LCS leakage control pathway is estimated to be

$$(0.9)(0.9)(0.99) = 0.802 \tag{C.5}$$

as shown in Figure C.2.

SJAE and Offgas System Pathway

The use of the steam jet air ejectors (SJAE) to reduce offsite releases maintains the main condenser vacuum. Operation of the SJAEs sweeps noncondensables to the offgas system. Any MSIV leakage would then be processed through the offgas system by opening the main turbine bypass valves or the steam line drain valves to direct MSIV leakage to the main condenser. Figure C.3 illustrates this leakage control pathway.

Once the material that leaks past the MSIVs is in the main condenser, the noncondensable radioactive gases are evacuated from the main condenser by the SJAE and processed through the offgas system recombiner, condenser, charcoal beds or delay tanks, and HEPA filters. Finally, they go out the main stack. This option provides the optimum treatment of the radioactive gases prior to release from the plant when there is no large steam line break. By using this pathway, any MSIV leakage is directed to the main condenser, thus precluding or minimizing any leakage from elsewhere in the main steam system.

This flow path also permits the cold-trapping of iodine and volatiles, and gives the added benefit of condensation, scrubbing, and plateout that would occur in the MSL and main condenser. Further plateout and holdup continues in the offgas system, the charcoal absorbers, and HEPA filters. This pathway also maximizes the holdup time of the radioactive gases and releases them from the plant at an elevated point.

The SJAEs use a high velocity jet of steam to create a low pressure for the removal of noncondensable gases from the main condenser shells. Main steam, reduced to 125 psia, is supplied through a strainer to each SJAE nozzle. The nozzle accelerates the steam to a high velocity so that it passes through the diffuser throat as it begins to expand. Gas molecules present in the suction chamber are entrained in and carried by the steam.

This control pathway relies on a flow path to the main condenser that is intact. For steam line break scenarios, this pathway becomes ineffective. For other scenarios involving radiation in the steam lines, the MSIVs are closed and the steam supply for the SJAEs is not available. Therefore, an auxiliary source of steam for operation of SJAEs and establishing turbine shaft seals is required. This steam source is assumed to be lost on LOOP, reflected by an additional 0.025 unavailability factor.

A success tree analysis of the availability of this leakage control pathway is illustrated in Figure C.4. Similar to the success tree presented

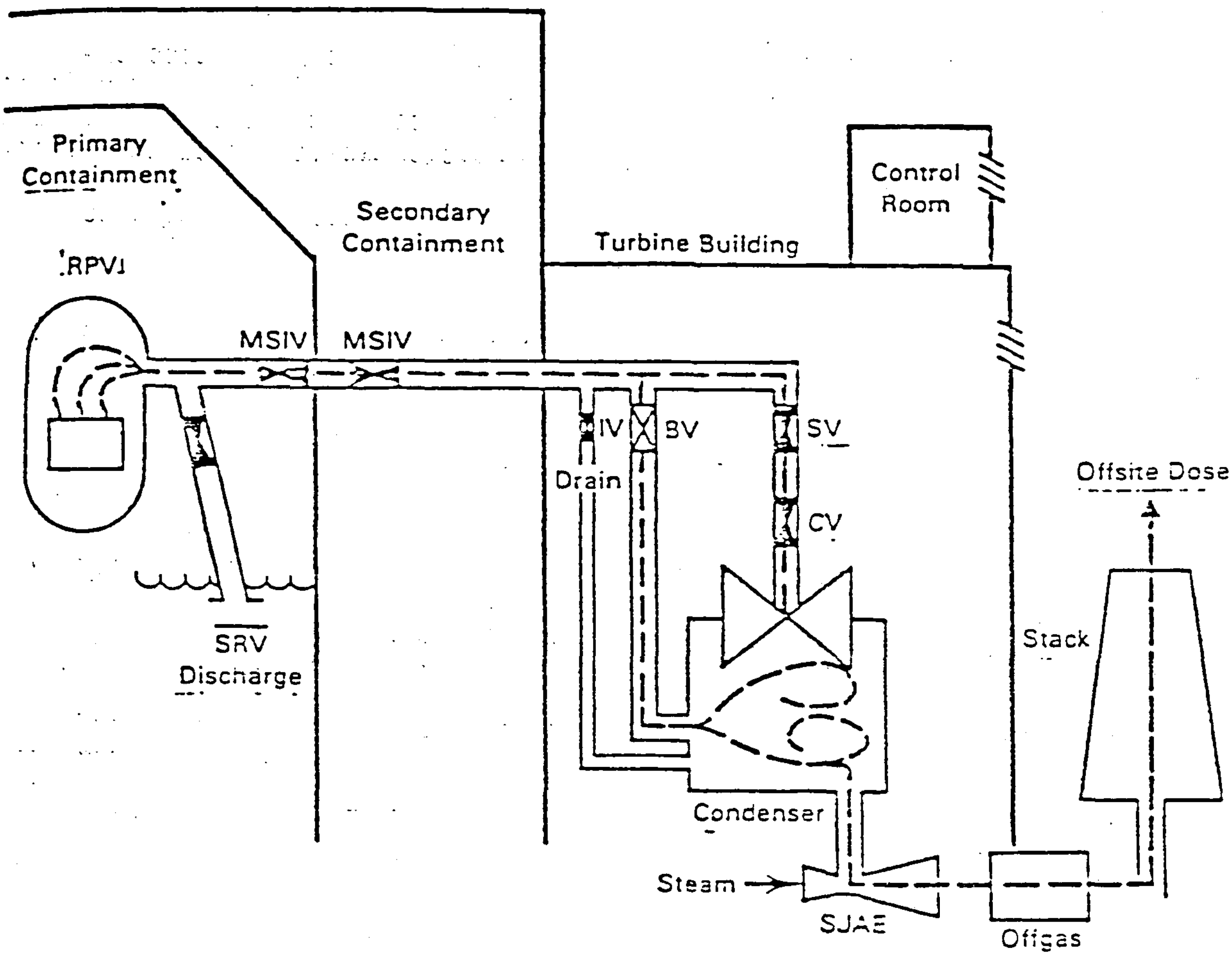

FIGURE C.3. SJAE and Offgas System Flow Path

in Figure C.2, there are four main conditions that must be met in Figure C.4 for the release via the SJAEs and offgas system to occur:

- operator section of this control pathway
- turbine bypass valve (TBV) opened, or main steam line drain isolation valves (IVs) open
- availability of steam supply and mechanical operation of SJAEs
- mechanical operation of the offgas treatment system.

The first condition requires that the operator select this control pathway. This requires opening the TBV or turbine stop valve (TSV) and aligning valves for delivery of the steam supply. Currently, this is not a standard emergency procedure, and the probability of this selection is negligible; however, the procedures could be modified to reflect this potential. A

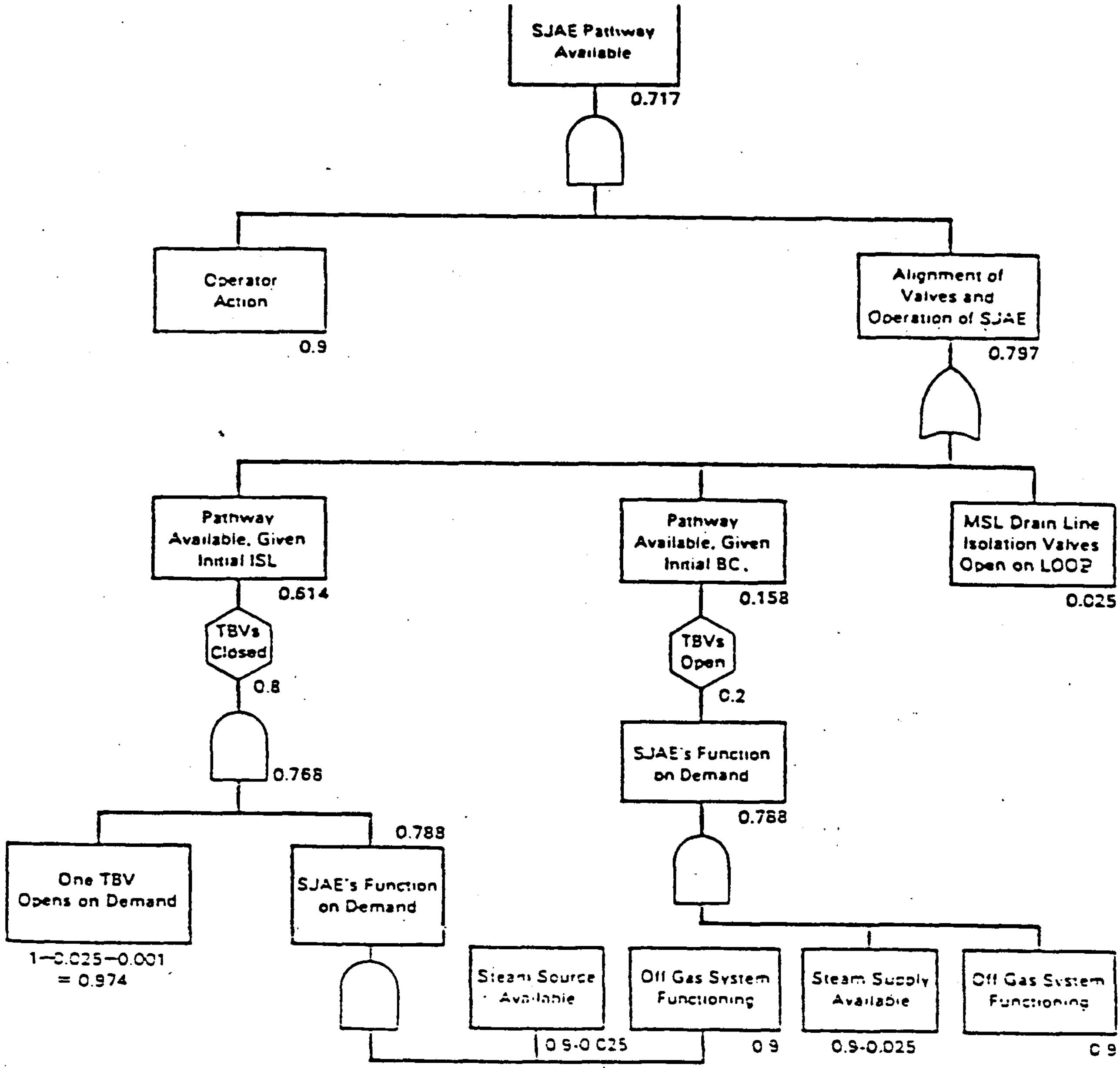

FIGURE C.4. SJAE and Offgas Availability Success Tree

probability of 0.9 is assumed if such procedures are implemented, consistent with the assumptions made in the MSIV-LCS analysis. Again, this value is assumed to include both the probability of operator action and the probability that the operator acts correctly once the decision is made.

In the second condition for using the SJAEs and offgas pathway, the TBV or main steam line drain line isolation valve must open on demand to open the pathway to the condenser. Depending on the initiating event that led to plant shutdown, the TBV could be in the open or closed position. For scenarios that open the TBV after shutdown, the TBV may remain open on closure of the MSIVs. No automatic closure signal is generated for the TBV on closure of the MSIVs. A review of NUREG-1169 (NRC 1986) for transients in the BWR 4 plant indicates that TBV opening could be expected for loss of off-site power and transients where the LCS is available. These contribute 2.92E-05/plant-year + 3.7E-06/plant-year out of a total frequency of 1.97E-04/plant-year, or about 20%. A similar percentage is obtained for the BWR 6. The LOCA scenarios are not thought to open the TBV, so probability of the TBV being open when the MSIVs close is estimated to be 20%. For 80% of the core-damage scenarios, the operator must then properly align the valve. As a result, the probability of the TBV failing to open on demand must be considered. This value is typically assumed to be about 1E-03/demand. Unavailability due to loss of offsite power of 0.025 is included here, for an availability of

$$1 - 0.001 - 0.025 \doteq 0.974 \tag{C.6}$$

The drain line isolation valve could also fail to open on LOOP. The potential for this was put at 0.025 for all shutdowns.

The third condition deals with the availability of steam for the SJAEs. Again, it is assumed that the plant in question has an auxiliary steam supply available. It is highly uncertain if adequate steam would be available from the auxiliary source after several hours into a core-damage accident. However, if no other systems are consuming steam, the availability may result from operator selection and alignment of the proper valves.

The probability of operator action to open the steam pathway is assumed to be synonymous with the overall decision to use this pathway, starting with the selection to open the TBV. Thus, a high probability of 0.8 is assumed for successful operator action. The mechanical success or failure of the SJAE function is then thought to result from the operation of valves that are used to isolate the SJAEs from the steam source. Several valves may be required to open to complete the steam pathway. To reflect the uncertainty in the steam supply and its maintenance after shutdown as well as the uncertainty in the valving and power supply arrangement, an availability value of 0.9 will be used.

For the fourth condition, availability of the offgas treatment system focuses on the operation of a blower because the rest of the components have basically a passive function. This availability is put at 0.9.

The last three conditions listed above for using the SJAE and offgas pathway deal with the proper alignment of valves. The basic discussion is the same as the one presented in the MSIV-LCS analysis. Based upon the values given in Figure C.4, the probability of proper valve alignment is estimated to be 0.797. Combined with operator interaction probability of 0.9, this will yield a total probability for the SJAE availability of 0.717.

Condenser Vacuum Pumps Pathway

The condenser vacuum pumps pathway is similar to the leakage control system pathway where the SJAEs are used to maintain a main condenser vacuum. In this option, mechanical vacuum pumps or turbine gland exhausters are used to maintain a slight condenser vacuum. Because use of the mechanical vacuum pumps is typically prohibited by technical specifications whenever significant concentrations of hydrogen may be present in the condenser, use of the gland exhausters is likely to be the only practical option.

This flow path is presented in Figure C.5. The MSIV leakage is drawn through the turbine bypass valves and into the main condenser. It is then pumped directly to the main stack. (In the SJAE and offgas system option, the leakage was processed through the offgas system before going through the main stack.)

Because all plants are equipped with mechanical vacuum pumps and gland exhausters, this leakage control pathway is an option for all plants. In this pathway, the radioactive gases are evacuated from the main condenser by the mechanical vacuum pumps or gland exhausters and pumped through a holdup volume and out the main stack for the plant. The mechanical vacuum pumps are used to remove air and noncondensables from the main condenser during startup when steam pressure is less than 260 psia, and they can operate as roughing pumps. They are typically vane-type centrifugal pumps, rated at 2350 scfm. Each is driven by a 100-hp electric motor.

The mechanical vacuum pumps isolate from the condenser on high steam line radiation. Direct operator action is required to initiate pump action when MSIV leakage control is needed. There is essentially no holdup time (about 2 minutes) and there are also no condensers, filters, or charcoal absorbers in the flow path to remove radioactive contaminants from the effluent.

This pathway is similar to the previous leakage control pathways; if there is a large steam line break (outside the capacity of the vacuum pumps), it becomes ineffective. A success tree analysis of the availability of this pathway is presented in Figure C.6. There are three conditions that need to be met before the vacuum pump/gland exhauster pathway is available:

- availability of the pathway
- mechanical operation of the vacuum pumps or gland exhausters
- proper alignment of valves.

This leakage control pathway requires that the TBVs or drain line isolation valves be open. As discussed in the analysis of the SJAEs and the offgas system, the initial plant condition following normal reactor trip is for the TBV to remain closed 80% of the time. The drain lines open only on LOOP.

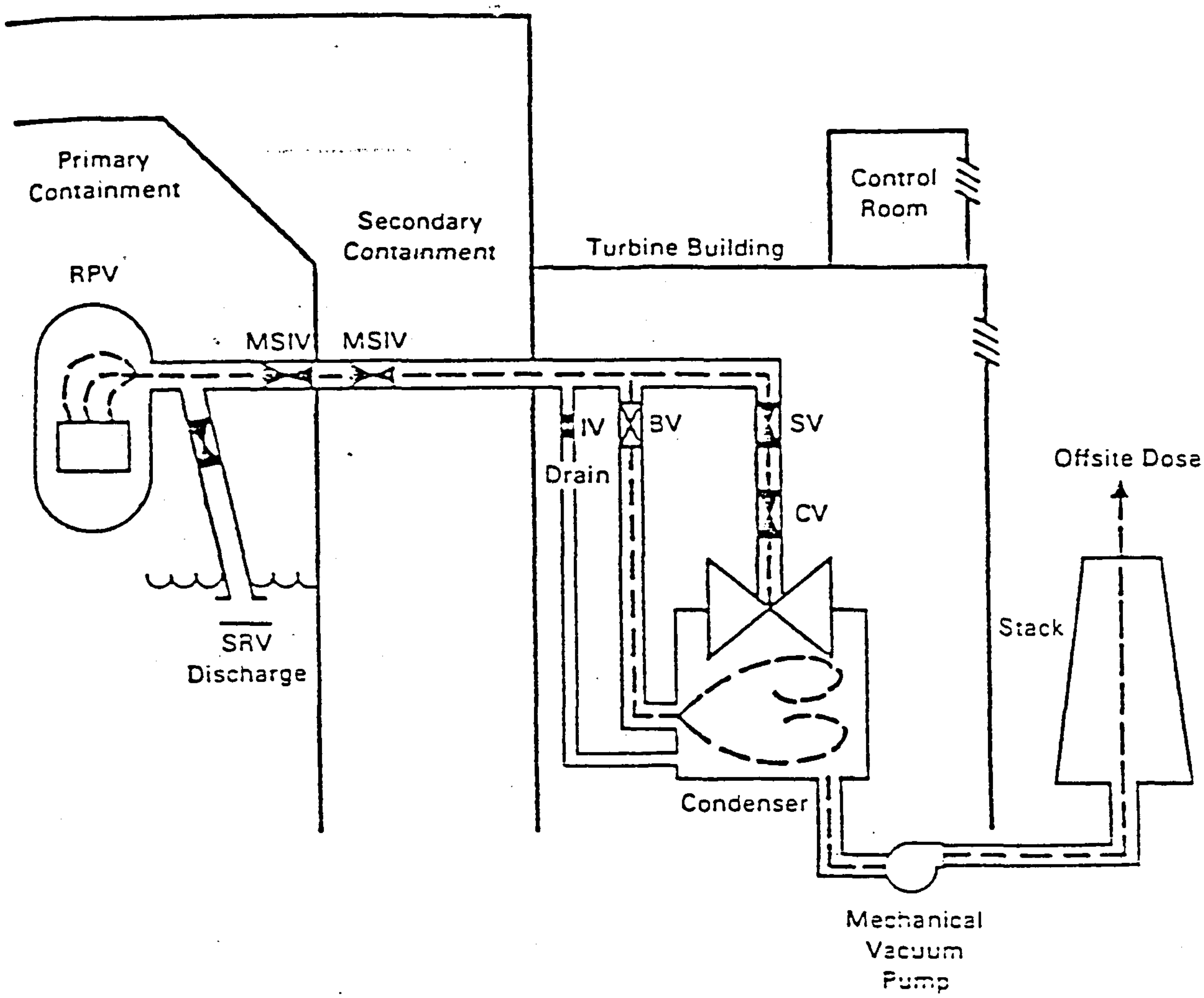

FIGURE C.5. Condenser Vacuum Pump (Gland Exhauster) Flow Path

With MSIV closure, the turbine control valve (TCV) closure also is assured. However, for some core damage scenarios, the initiating events could result in turbine trip and TCV closure before any demand signal for MSIV closure. In such cases, the TBV opens to bleed steam pressure directly to the condenser, and the pathway already is aligned properly. Rather than make a detailed study of scenarios, it is assumed that the TBV is open with MSIVs closed in 20% of the cases. As a result, the operator has to decide to use this leakage control pathway, requiring the TBV and vacuum pumps. This is shown in Figure C.6 for the availability of the pathway. The discussion for proper valve alignment is similar to previous release pathways. A probability of failure to open on demand for the TBV valve is 1E-03. The same value typically is used for pumps. Electric motors have a slightly lower value for failure to start of 3E-04/demand and failure to run of 1E-05/hr to 1E-03/hr for extreme environments. The addition of power supplies and switching for the motors make a make a 0.9 value for availability (as used for the SGTS system) more reasonable. Unavailability factors of 0.025 due to LOOP are also added to both components.

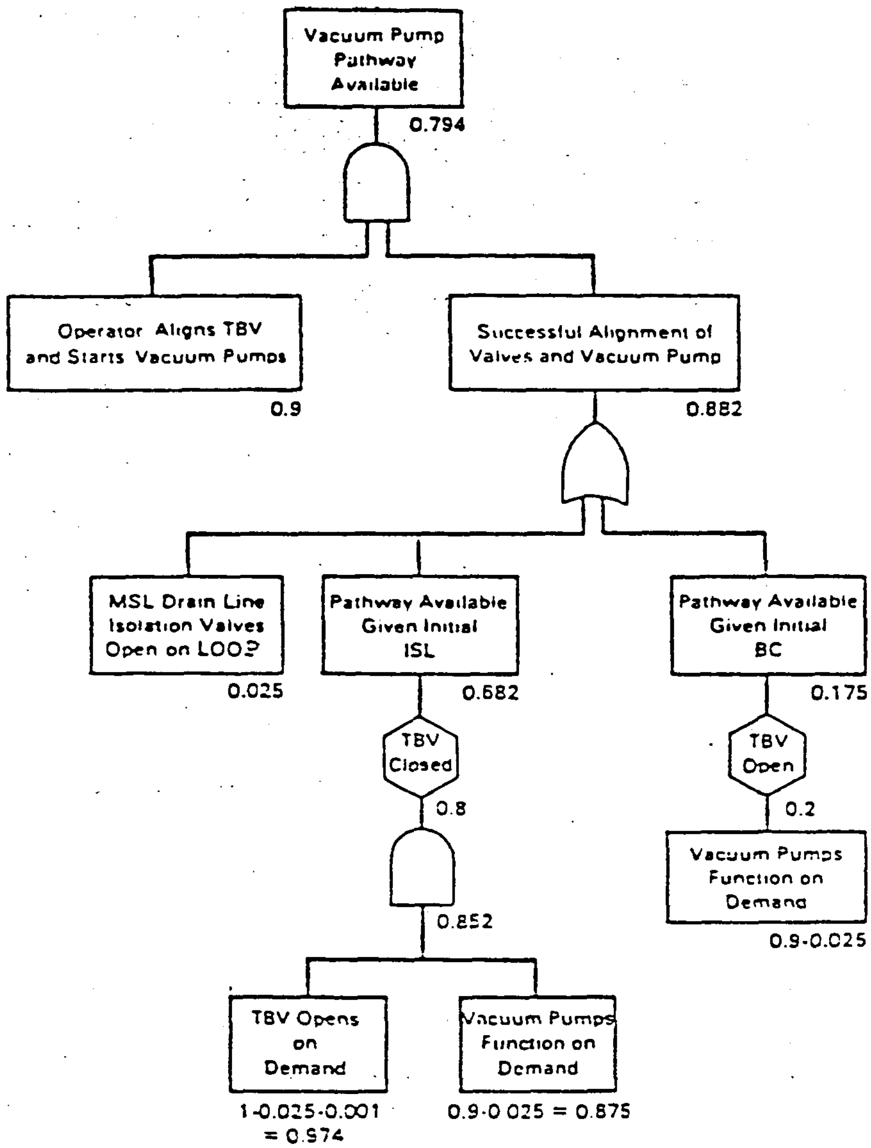

FIGURE C.6. Condenser Vacuum Pump (Gland Exhauster) Availability Success Tree

Applying the data given in Figure C.6 yields a total probability for the availability of mechanical vacuum pumps of 0.794.

Containment Within the Main Steam System Pathway

This leakage control option isolates the main steam system by closing the main turbine stop, control, bypass valves, and all branch line valves that connected to the MSL. Figure C.7 illustrates this leakage control pathway. The TBV is assumed to be closed in 80% and open in 20% of the

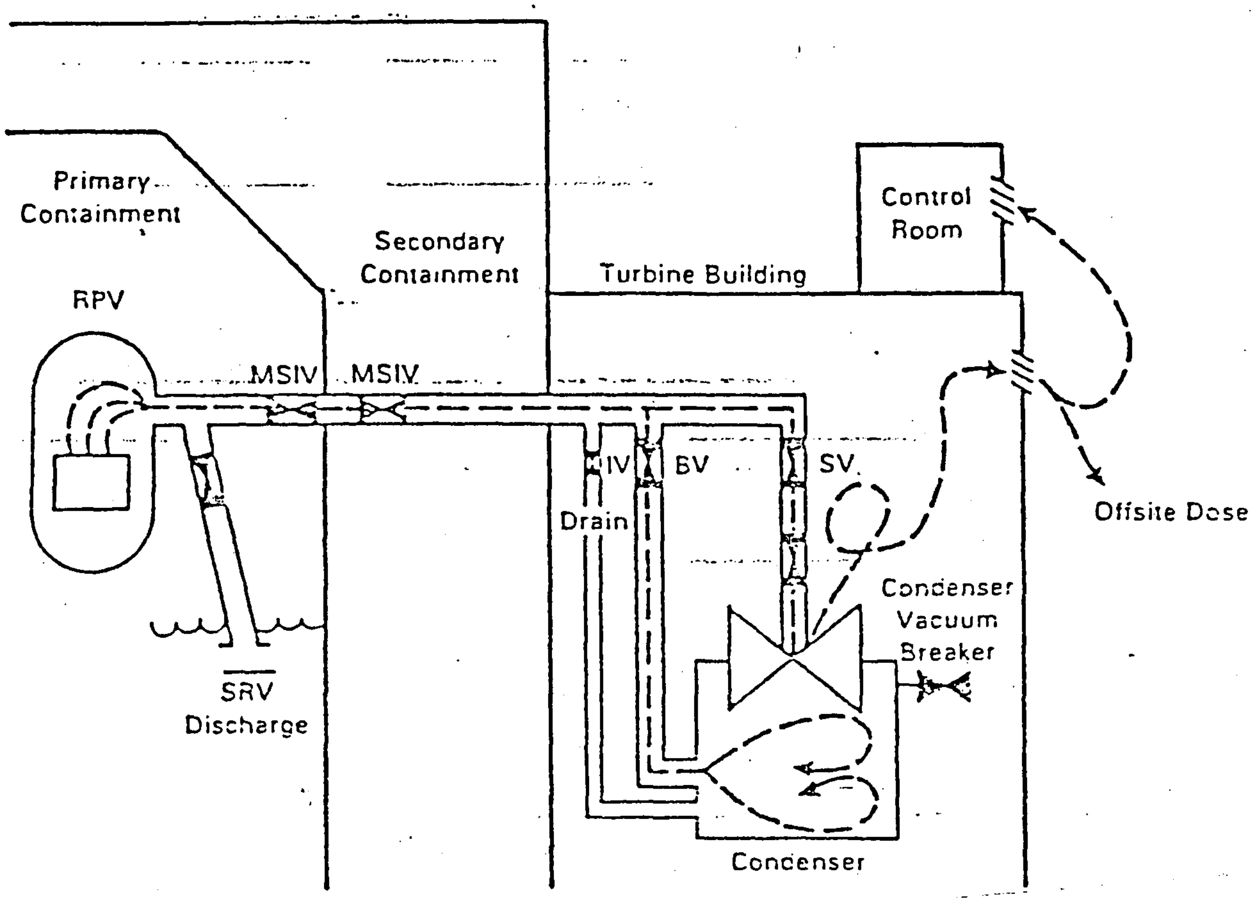

FIGURE C.7. Isolated Steam Line Flow

scenarios, with the latter requiring operator action to close the valve. A 0.9 value for operator choice of action and successful completion of that action is used to allow direct comparison to other pathways.

The net availability estimated in Figure C.8 is 0.962.

To contain any possible leakage past the main turbine stop and control valves, the main condenser vacuum breakers are closed to isolate the main condenser. Turbine seals are established with an auxiliary steam source. This pathway then essentially becomes the isolated condenser pathway discussed in the next section.

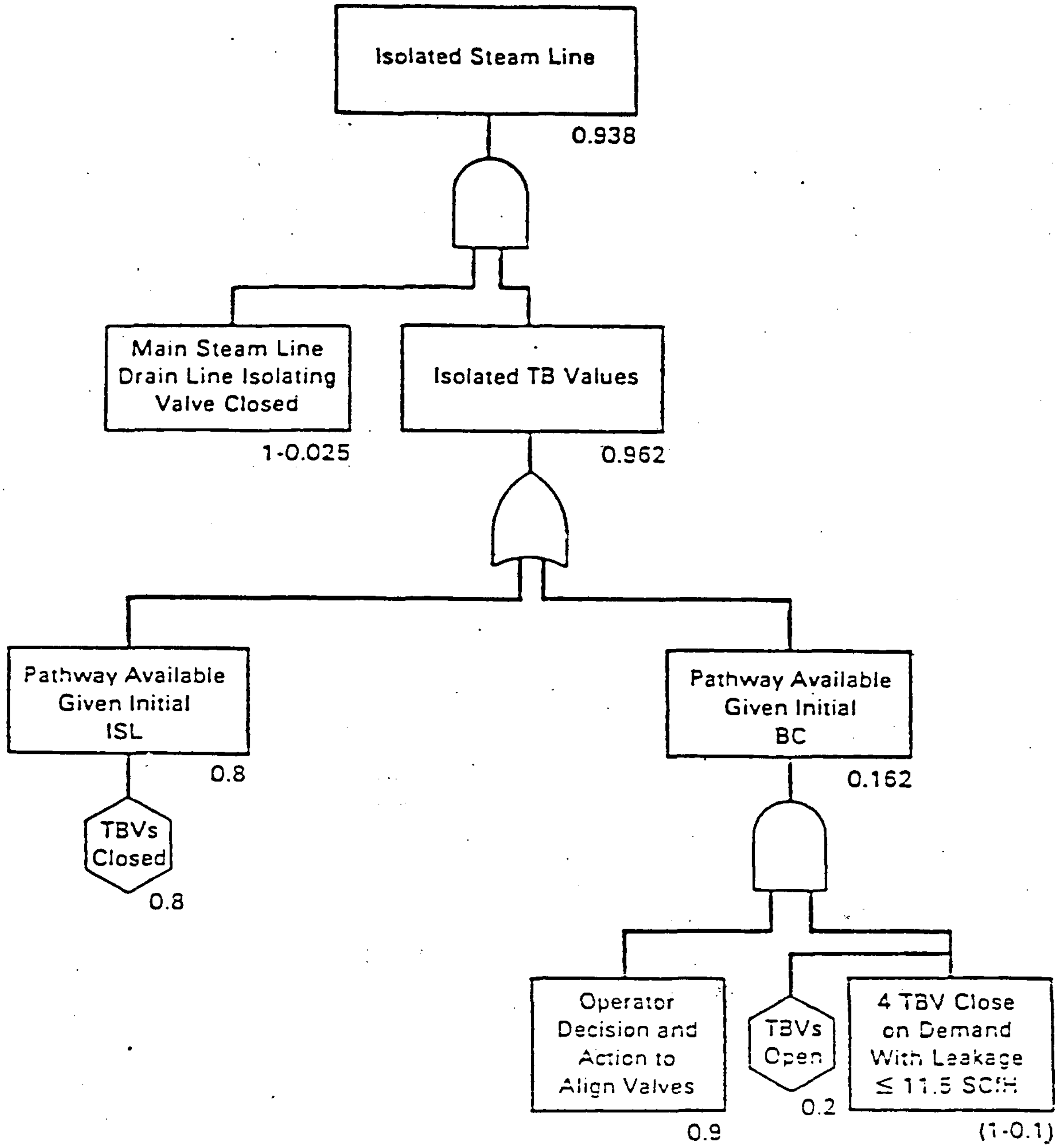

FIGURE C.8. Isolated Steam Line Flow Path Availability Success Tree

Isolated Condenser Pathway

In this leakage control pathway, illustrated in Figure C.9, MSIV leakage is directed to a sealed or isolated condenser. Leakage into the secondary containment and treatment by the SGTS system are not considered. Because the leakage from the turbine building depends on its leakage constraints, any leakage from the turbine building would constitute a direct, unfiltered release from the plant.

This flow path allows the cold-trapping of iodine and volatile solids, plus the scrubbing and plateout that occurs in the MSL and main condenser (provided TBVs or the main steam line drain line isolation valves are open).

Containment within the turbine building provides some holdup prior to release from the plant. This might be the only release control pathway for radioactive gases if there is a steam line break outside the primary containment.

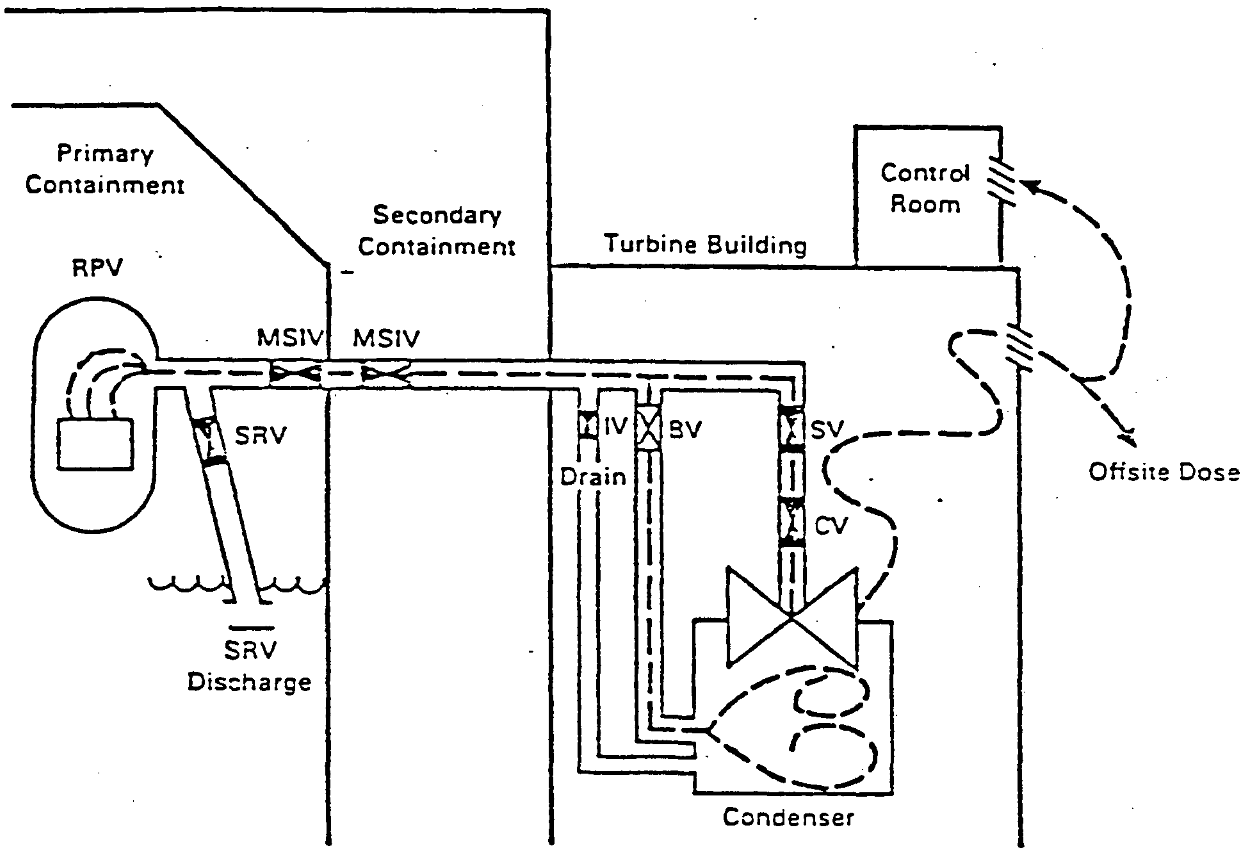

FIGURE C.9. Isolated Condenser Flow Path

A success tree analysis of the availability of this leakage control pathway is presented in Figure C.10. Operator action with a probability of 0.9 to open TBVs is assumed as before to allow direct comparisons between the options. This leakage control pathway requires that the TBVs or drain line isolation valves be open. As discussed in the analysis of the SJAE and offgas system option, the initial plant condition following normal reactor trip is for the TBV to remain closed 80% of the time. The drain lines open only on LOOP. The net result from Figure C.10 is an estimated availability of the isolated condenser pathway of 0.926.

Summary of MSIV Leakage Control Pathways

The estimated availability of the five leakage control pathways is summarized in Table C.8. The exact configuration of these systems has a pronounced effect on the estimates, as does the assumed initial condition of loss of offsite power. The operation of all of the leakage control pathways, however, relies on relatively simple mechanical components, such as blowers, valves, switches, and power supplies.

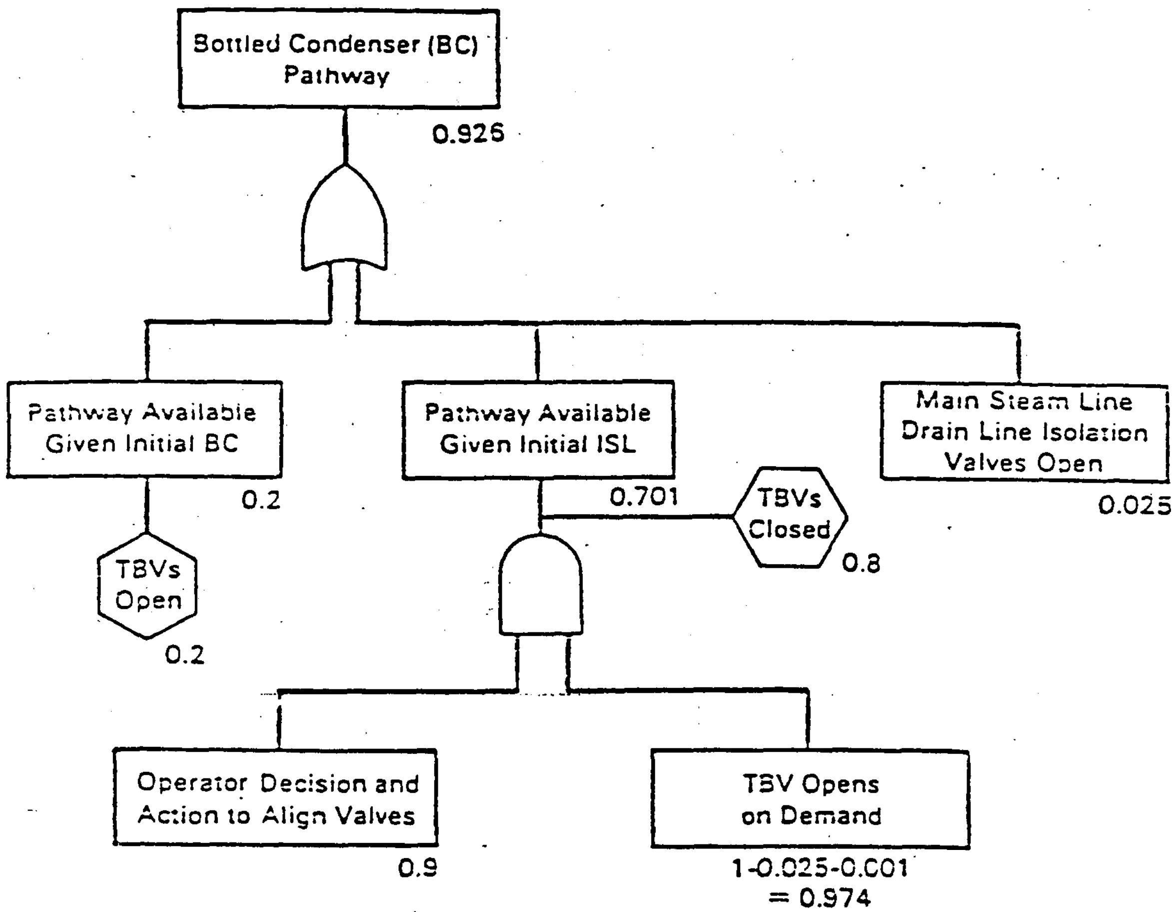

FIGURE C.10. Isolated Condenser Availability Success Tree

TABLE C.7. Summary of Availability of Leakage Pathways

	Probability of Configuration on Plant Shutdown	Net Availability with Operator Action
Leakage control system (LCS)	--	0.802
Steam jet air ejectors (SJAE) and offgas system	--	0.717
Condenser vacuum pumps	--	0.794
Isolated main steam system	0.775	0.938
Isolated condenser (no HVAC required)	0.225	0.926
Total	1.0	

Again, it was assumed that the plant would configure itself with closed TBVs 80% of the time. This would be lowered slightly by the potential for steam line drain line isolation valves opening on LOOP (0.025). Without any operator action, the plant is expected to be in the isolated steam line configuration for (0.80 - 0.025), or 78% of the time, and in the isolated condenser configuration for 22% of the time. If operator action is considered, the potential availability of all pathways improve, but the isolated steam line configuration still has the highest availability estimate.

The potential for equipment unavailability due to loss of offsite power was again included in these estimates, but in a probabilistic sense. The probability of LOOP on shutdown without recovery within 30 minutes was put at 0.025. The reliance on these pathways for a safety grade leakage control option would thus require assumed power supplies in addition to procedural changes. The availability of these pathways could then be expected to improve, but only slightly.

After shutdown with the MSIVs closed, the plant is expected to be in either the isolated condenser or isolated steam line condition. The above analysis considers the potential for realignment of valves by the operator, and the availability of equipment necessary for the other pathways.

EFFECTIVE FREQUENCY OF PUBLIC DOSE

This section presents the correlation between person-rem of public dose per core-damage event and frequency of the event, giving an estimate of the person-rem/plant-year that is expected for this issue.

The dose consequence modeling effort presented in NUREG-1169 (NRC 1986) included a calculation of total whole body dose commitment to the public as a result of MSIV leakage. These calculations assumed a population density of 340 people per square mile out to 50 miles, the same value used in prioritization evaluations of safety issues. The estimated whole body public exposure as a function of MSIV leakage rate is summarized in Table C.9, along with the estimate of the frequency of such releases. Public exposure for each of several MSIV leakage rates was calculated. Estimates of public risk then were calculated using the core-damage frequency values cited in Appendix B.

Incorporated into Table C.9 are the availabilities of the leakage control pathways, which were summarized in Table C.8. The system is expected to be in either an isolated steam line or isolated condenser configuration after shutdown.

The probability-weighted public dose per event is the sum of the dose for each leakage rate via a particular pathway times the availability of the pathway at that leak rate. For the isolated condenser via the TBV or condensate drains, which have an availability of 0.93, the balance of the availability (1 - 0.93, or 0.07) is assumed to be the less-effective isolated main steam line case. Thus for the internal and nonseismic induced core damage accidents, for which the steam lines and condenser are considered to be usable, the public risk from leakage at 11.5 scfh per line (46 scfh total) can be bounded as follows:

$$\begin{aligned}\begin{array}{l}\text{Public risk}\\\text{(person-rem/}\\\text{plant-year)}\end{array} = &\left[\begin{array}{l}\text{core damage}\\\text{probability}\\\text{(internal \&}\\\text{nonseismic)}\\\text{per plant-yr}\end{array}\right]\left[\left(\begin{array}{l}\text{consequence of}\\\text{leakage at 46}\\\text{scfh via iso.}\\\text{condenser and}\\\text{TBV}\end{array}\right)\left(\begin{array}{l}\text{Availability}\\\text{iso. condenser}\\\text{\& TBV path}\end{array}\right)\right.\\ &\left. + \left(\begin{array}{l}\text{Consequence}\\\text{of leakage}\\\text{at 46 scfh}\\\text{via iso.}\\\text{steam line}\end{array}\right)\left(\begin{array}{l}\text{1 - Availability}\\\text{of oso. conden-}\\\text{ser \& TBV path}\end{array}\right)\right]\left[\begin{array}{l}\text{Probability of}\\\text{total leakage}\\\text{at <46 scfh}\end{array}\right]\end{aligned} \tag{C.6}$$

For the BWR 2/3/4, the public risk is:

$$\begin{aligned}\begin{array}{l}\text{Public risk}\\\text{(46 scfh)}\end{array} &= [2.0\text{E-4/plant-yr}][(6.4\text{E+0 person-rem})(0.93)\\ &\quad + (3.5\text{E+1 person-rem})(0.07)][0.91117]\\ &= 1.5\text{E-3 person-rem/plant-yr}\end{aligned} \tag{C.7}$$

For the leakage range from 46 to 400 scfh, the bounding risk is:

[2.0E-4/plant-yr][(6.0E+1 person-rem)(0.93)

+ (1.3E+3 person-rem)(0.07)][0.08637]

= 2.5E-3 person-rem/plant-yr (C.8)

For the leakage range from 400 to 4000 scfh, the bounding risk is:

[2.0E-4/plant-yr][(4.8E+2 person-rem)(0.93)

+ (2.8E+4 person-rem)(0.07)][0.00246]

= 1.2E-3 person-rem/plant-yr (C.9)

Total public risk for all leakage rates, internal and nonseismic events, isolated condenser is:

(1.5E-3 person-rem/plant-yr)

+ (2.5E-3 person-rem (plant-yr)

+ (1.2E-3 person-rem/plant-yr)

= 5.2E-3 person-rem/plant-yr (C.10)

For seismic-event-induced core-damage events, the steam lines and condenser are assumed to be unavailable. Therefore, the release is via the direct release path with no holdup or filtration. As above, the bounding risk from this family of core damage events is:

Public risk (0 to 46 scfh) = [8.0E-5/plant-yr]

[(1.2E+4 person-rem)(1)][0.91117]

= 8.7E-1 person-rem/plant-yr (C.11)

Public risk (46 to 400 scfh) = [8.0E-5/plant-yr]

[(1.04E+5 person-rem)(1)][0.08637]

= 7.2E-1 person-rem/plant-yr (C.12)

Public risk (400 to 4000 scfh) = [8.0E-5/plant-yr]

[(1.04E+6 person-rem)(1)][0.00246]

= 2.0E-1 person-rem/plant-yr (C.13)

Total public risk for all leakage rates, seismic-event-induced core-damage accidents, direct pathway

$$= 1.8E0 \text{ person-rem/plant-yr} \tag{C.14}$$

For BWR 2/3/4 total public risk, all core-damage causes, all leakage rates, isolated condenser pathway

$$= 1.8E0 + 5.2E{-}3 = 1.8E0 \text{ person-rem/plant-yr} \tag{C.15}$$

The constituents of the risk totals vary for the BWR 5/6 only by the ratio of the core damage accident frequencies. Thus, for the BWR 5/6:

Public risk for all leakage rates, internal and nonseismic events, isolated condenser pathway

$$= \frac{3.7E{-}5/\text{plant-yr}}{2.0E{-}4/\text{plant/yr}} (5.2E{-}3 \text{ person-rem/plant-yr})$$

$$= 9.6E{-}4 \text{ person-rem/plant-yr} \tag{C.16}$$

Public risk for all leakage rates, seismic event-induced core-damage accidents direct pathway

$$= \frac{6.0E{-}7/\text{plant-yr}}{8.0E{-}5/\text{plant-yr}} (1.8E0 \text{ person-rem/plant-yr})$$

$$= 1.4E{-}2 \text{ person-rem/plant-yr} \tag{C.17}$$

BWR 5/6 total public risk, all core-damage causes, all leakage rates, isolated condenser pathway

$$= (1.4E{-}2) + (9.6E{-}4) = 1.4E{-}2 \text{ person-rem/plant/yr} \tag{C.18}$$

For the case of the isolated main steam line, which has an availability of 0.94, the balance of the availability (1 - 0.94) is made up of the pathways that would exist if the actions to isolate the steam line were not taken or were unsuccessful. In either of these cases, the resulting configuration is essentially the same as for the isolated condenser, which is more effective in holding up and attenuating any MSIV leakage. Therefore, an availability of 1 will be used to express the isolated steam line characteristics (person-rem per event).

For BWR 2/3/4, the bounding public risk due to internal and nonseismic core-damage accidents for the isolated main steam line pathway is:

Public risk (46 scfh) = [2.0E-4/plant-yr]

[(3.5E+1 person-rem)(1)][0.91117]

= 6.4E-3 person-rem/plant-yr (C.19)

Public risk (400 scfh) = [2.0E-4/plant-yr]

[(1.3E+3 person-rem)(1)][0.08637]

= 2.2E-2 person-rem/plant-yr (C.20)

Public risk (4000 scfh) = [2.0E-4/plant-yr]

[(2.8+4 person-rem)(1)][0.00246]

= 1.4E-2 person-rem/plant-yr (C.21)

Total public risk for all leakage rates, internal

and nonseismic events, isolated steam line

= 4.2E-2 person-rem/plant-yr (C.22)

For the seismic-event-induced core-damage accidents, the nonseismic Category 1 steam lines are assumed to fail and the release consequence is calculated as though the release is direct to atmosphere at ground level. As calculated for the isolated condenser case, this risk value is 1.8E0 person-rem/plant-yr for BWR 2/3/4. The total public risk for implementation of the isolated steam line strategy for a BWR 2/3/4 is (4.2E-2) + (1.8E0), or 1.8E0 person-rem/plant-yr.

Applying the BWR 5/6 core damage probabilities to the previously calculated values for risk per event, the whole body public exposure per BWR 5/6 plant-yr presented in Table C.9 were determined.

The mechanical vacuum (gland exhauster) strategy is shown in Figure C.5 to have an availability of 0.80. The balance of the availability (0.20) is made up of those paths that would exist if operator actions to initiate the mechanical vacuum pump pathway were not taken or were unsuccessful (i.e., isolated steam line or isolated condenser). Since the largest part of the availability decrement is assumed to be operator-related, it will be assumed that the balance of the availability will be the more passive of the two (requiring the least operator intervention to accomplish), that is the isolated steam line.

TABLE C.9. Summary of BWR Public Exposures

Core Damage Cause and Pathway Scenario	Availability of Pathway	Total Leak Rate, scfh	Person-rem per Event(a)	Whole Body Public Exposure, person-rem/plant-yr BWR 4	BWR 6
Isolated condenser					
Internal & non-seismic-induced core-damage accidents	0.93	<46	7.7E0	1.5E-3	2.8E-4
		46-400	1.3E+1	2.5E-3	4.7E-4
		400-4000	5.9E0	1.2E-3	2.2E-4
Seismic-induced core-damage accidents	0	<46	1.1E+4	8.7E-1	6.6E-3
		46-400	9.0E+3	7.2E-1	5.4E-3
		400-4000	2.6E+3	2.0E-1	1.6E-3
Risk total				1.8E0	1.4E-2
Isolated main steam line					
Internal & non-seismic-induced core-melts	0.94	<46	3.2E+1	6.4E-3	1.2E-3
		46-400	1.1E+2	2.2E-2	4.1E-3
		400-4000	6.9E+1	1.4E-2	2.6E-3
Seismic-induced core-melts	0	<46	1.1E+4	8.7E-1	6.6E-3
		46-400	9.0E+3	7.2E-1	5.4E-3
		400-4000	2.6E+3	2.0E-2	1.6E-3
Risk total				1.8E0	2.2E-2
Mechanical vacuum pump (gland exhauster)					
Internal & non-seismic-induced core-melts	0.8	<46	1.1E+1	2.2E-3	4.0E-4
		46-400	2.6E+1	5.2E-3	9.7E-4
		400-4000	1.5E+1	2.9E-3	5.4E-4
Seismic-induced core-melts	0	<46	1.1E+4	8.7E-1	6.6E-3
		46-400	9.0E+3	7.2E-1	5.4E-3
		400-4000	2.6E+3	2.0E-2	1.6E-3
Risk total				1.8E0	1.6E-2
Steam jet air ejector - continuous Operation					
Internal & non-seismic-induced core-melts	0.72	<46	9.3E0	1.9E-3	3.4E-4
		46-400	3.2E+1	6.3E-3	1.2E-4
		400-4000	1.9E+1	3.9E-3	7.2E-4
Seismic-induced core-melts	0	<46	1.1E+4	8.7E-1	6.6E-3
		46-400	9.0E+3	7.2E-1	5.4E-3
		400-4000	2.6E+3	2.0E-2	1.6E-3
Risk total				1.6E0	1.5E-2
Leakage control system					
Internal & non-seismic-induced core-damage accidents	0.8	<46	8.7E+1	1.7E-2	3.2E-3
		46-400	8.9E+1	1.8E-2	3.3E-3
		400-4000	6.9E+1	1.4E-2	2.5E-3
Seismic-induced core-melts	0.8	<46	8.7E+1	6.9E-3	5.2E-5
		46-400	8.9E+1	7.2E-3	5.3E-5
		400-4000	6.9E+1	5.5E-3	4.1E-5
Risk total				6.9E-2	9.1E-3

(a) These values are weighted for availability of the specified pathway. This table represents the leak-rate-weighted person-rem per core-damage event if the specified pathway has been implemented and is available at the specified level.

For BWR 2/3/4, the bounding public risk due to internal and nonseismic-event-induced core-damage accidents for the mechanical vacuum pump pathway is:

$$\text{Public risk (46 scfh)} = [2.0E\text{-}4/\text{plant-yr}]$$
$$[(6.1E0 \text{ person-rem})(0.8) + (3.5E+1)(0.2)][0.91117]$$
$$= 2.2E\text{-}3 \text{ person-rem/plant-yr} \quad (C.23)$$

$$\text{Public risk (400 scfh)} = [2.0E\text{-}4/\text{plant-yr}]$$
$$[(5.3E+1 \text{ person-rem})(0.8) + (1.3E+3)(0.2)][0.08637]$$
$$= 5.2E\text{-}3 \text{ person-rem/plant-yr} \quad (C.24)$$

$$\text{Public risk (4000 scfh)} = [2.0E\text{-}4/\text{plant-yr}]$$
$$[(4.3E+2 \text{ person-rem})(0.8) + (2.8E+4)(0.2)][0.00246]$$
$$= 2.9E\text{-}3 \text{ person-rem/plant-yr} \quad (C.25)$$

Applying the method previously demonstrated for the seismic-event-induced core-damage accidents and the BWR 5/6 core-damage frequencies, the public risk values presented in Table C.9 were calculated.

For the steam jet air ejector pathway, which has an availability of 0.72, the balance of the availability is assumed to be made up of the isolated steam line pathway which would exist if no operator action were taken to open the turbine bypass or steam line drain valves.

For BWR 2/3/4, the public risk due to internal and non-seismic core damage accidents for the SJAE/offgas system pathway maintained in continuous operation is:

$$\text{Public risk (46 scfh)} = [2.0E\text{-}4/\text{plant-yr}]$$
$$[(5.4E\text{-}1 \text{ person-rem})(0.72) + (3.5E+1)(0.28)][0.91117]$$
$$= 1.9E\text{-}3 \text{ person-rem/plant-yr} \quad (C.26)$$

$$\text{Public risk (400 scfh)} = [2.0E\text{-}4/\text{plant-yr}]$$
$$[(4.7E0 \text{ person-rem})(0.72) + (1.3E+3)(0.28)][0.08637]$$
$$= 6.3E\text{-}3 \text{ person-rem/plant-yr} \quad (C.27)$$

Public risk (4000 scfh) = [2.0E-4/plant-yr]

[(4.7E1 person-rem)(0.72) + (2.8E+4)(0.28)][0.00246]

= 3.9E-3 person-rem/plant-yr (C.28)

Applying the method previously demonstrated for the seismic-event-induced core-damage accidents and the BWR 5/6 core-damage frequencies, the public risk values presented in Table C.9 were calculated.

For the leakage control system pathway which has an availability of 0.80, the balance of the availability is assumed to be made up of the isolated main steam line pathway. For seismic-event-induced core-damage accidents, the LCS is assumed to remain operable. For total leak rates in excess of 400 scfh, the LCS is presumed to be nonfunctional and the release is by way of the isolated steam line for the internal/nonseismic core-damage accidents and direct to atmosphere for the seismically induced core-damage accidents.

For BWR 2/3/4, the bounding public risk due to internal and nonseismic core-damage accidents for the LCS pathway is:

Public risk (46 scfh) = [2.0E-4/plant-yr]

[(1.1E+2 person-rem)(0.80) + (3.5E+1)(0.20)][0.91117]

= 1.7E-2 person-rem/plant-yr (C.29)

Public risk (400 scfh) = [2.0E-4/plant-yr]

[(9.7E+2 person-rem)(0.80) + (1.3E+3)(0.2)][0.08637]

= 1.8E-2 person-rem/plant-yr (C.30)

Public risk (4000 scfh) = [2.0E-4/plant-yr]

[(2.8E+4 person-rem)(1)][0.00246]

= 1.4E-3 person-rem/plant-yr (C.31)

Again, using the method previously demonstrated for seismic-event-induced core-damage accidents and the BWR 5/6 core damage frequencies, the public risk values presented in Table C.9 were calculated.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report describes the analysis conducted to establish the basis for answering two remaining regulatory questions facing the NRC staff regarding the resolution of Generic Issue C-8, specifically: 1) What action should the NRC take concerning plants that currently have a leakage control system (LCS)? 2) What action should the NRC take concerning plants that do not have an LCS? Using individual MSIV leak test data, the performance of a system of eight such valves in a standard BWR configuration was modeled. The performance model was used along with estimates of core damage accident frequency and calculated dose consequences to determine the public risk associated with each of the alternatives. The occupational exposure implications of each alternative were calculated using estimates of labor hours in radiation zones that would be incurred or avoided. The costs to industry of implementing each alternative were estimated using standard cost formulae and NRC staff estimates. The costs to the NRC were estimated based on the effort incurred or avoided for reviews or other staff actions engendered by the selection of a particular alternative. The costs and risks thus calculated suggest that no regulatory action can be justified on the basis of risk reduction or cost savings.

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