NUREG–1372

Regulatory Analysis for the Resolution of Generic Issue C–8, "Main Steam Isolation Valve Leakage and LCS Failure"

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

C. C. Graves

AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 2120 L Street, NW, Lower Level, Washington, DC 20555
2. The Superintendent of Documents, U.S. Government Printing Office, P.O. Box 37082, Washington, DC 20013-7082
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the Code of Federal Regulations, and Nuclear Regulatory Commission Issuances.

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. Federal Register notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Information Resources Management, Distribution Section, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

NUREG--1372

TI90 012677

Regulatory Analysis for the Resolution of Generic Issue C–8, “Main Steam Isolation Valve Leakage and LCS Failure”

Manuscript Completed: August 1989
Date Published: June 1990

C. C. Graves

Division of Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ABSTRACT

Generic Issue C–8 deals with staff concerns about public risk because of the incidence of leak test failures reported for main steam isolation valves (MSIVs) at boiling water reactors and the limitations of the leakage control systems (LCSs) for mitigating the consequences of leakage from these valves. If the MSIV leakage is greatly in excess of the allowable value in the technical specifications, the LCS would be unavailable because of design limitations. The issue was initiated in 1983 to assess (1) the causes of MSIV leakage failures, (2) the effectiveness of the LCS and alternative mitigation paths, and (3) the need for additional regulatory action to reduce public risk. This report presents the regulatory analysis for Generic Issue C–8 and concludes that no new regulatory requirements are warranted.

CONTENTS

FIGURES

TABLES

EXECUTIVE SUMMARY

Generic Issue C–8 deals with staff concerns about public risk resulting from excessive leakage from main steam isolation valves (MSIVs) at boiling water reactors and limitations of the leakage control systems (LCSs) provided to mitigate the consequences of this leakage. Because of excessive leakage problems encountered with these valves and staff concerns that the leakage would compromise the containment function, the staff formulated a position to require installation of a safety-grade LCS on all BWRs with construction permits issued after March 1, 1970. In the late 1970s and early 1980s, staff concerns increased because of the incidence of high MSIV leak rates. Because of LCS design limitations, the system would be unavailable if the leak rate were greatly in excess of the allowable value in the technical specifications. Hence, Generic Issue C–8 was initiated in 1983 to assess (1) the causes of MSIV leakage failures, (2) the effectiveness of the LCS and alternative mitigation paths, and (3) the need for regulatory action to limit public risk. Independently, the BWR Owners Group (BWROG) formed a committee to assess the MSIV leakage problem and develop recommendations for reducing the leakage.

The efforts of the staff and its contractor, Pacific Northwest Laboratories (PNL), through 1986 were covered in NUREG–1169, a copy of which was sent to all licensees. This report concluded that there was a high likelihood that the BWROG had identified the key causes of high MSIV leakage. The NUREG–1169 calculations showed several non-Seismic Category I paths that gave lower offsite doses than the LCS. The risk analysis at that time, based on estimates of improved MSIV leakage and nonseismic events, indicated low public risk for all paths. However, these results and the resolution of Generic Issue C–8 were limited by the lack of confirmatory leakage data to support the BWROG recommendations on corrective actions. In August 1988, the BWROG supplied results of a new survey, which showed a significant reduction in MSIV leakage. The survey, which involved 329 test points from 24 BWRs, confirms the earlier conclusions that the key reasons for high leakage had been identified and is considered to represent the current capabilities of the whole BWR plant population. PNL updated the analyses of NUREG–1169 to incorporate the new BWROG leakage data and to include the effects of seismic events in the assessment of public risk due to MSIV leakage with and without an LCS. Although relatively small risk values were obtained, additional calculations were performed to assess the cost-effectiveness of various alternative actions.

The following alternatives were considered:

Alternative 1—No Action.

Alternative 2—Require addition of standard capacity LCS to plants without an LCS.

Alternative 3—Require increased capacity LCS on all plants.

Alternative 4—Disable currently installed LCSs.

Alternative 5—Use other mitigation paths with larger decontamination factors.

On the basis of a value-impact analysis, it was concluded that no backfit requirements to reduce public risk associated with MSIV leakage or the LCS are warranted and that Alternative 1 should be adopted. This alternative maintains the current requirements, systems, and leakage treatment practices. All licensees would be expected to continue their efforts to maintain the LCS and satisfactory MSIV performance. This alternative does not preclude a licensee from proposing an alternative action based on plant-specific considerations.

1. STATEMENT OF PROBLEM

1.1 Introduction

Generic Issue C-8 deals with staff concerns about public risk from excessive leakage of main steam isolation valves (MSIVs) at boiling water reactors (BWRs) and limitations of the leakage control systems (LCSs) for these valves that were installed at later plants. BWRs use two Y-pattern globe valves in series to isolate each main steam line from the non-nuclear balance of plant. These MSIVs serve to protect against a steam line break outside containment and limit the release of fission products to the environment in the event of a core damage incident. Hence, rather stringent leakage limits were applied to these valves for all BWRs.

During the early operating history of BWRs in the 1970s, a large number of MSIV leak test failures were reported. Because of staff concerns that this leakage could compromise the containment function, the staff formulated a position to require installation of safety-grade LCSs to treat this leakage on all BWRs with construction permits issued after March 1, 1970. In the late 1970s and early 1980s, the staff concern increased because of more reports of very high leak rates. Because of design limitations, the LCS would be unavailable if the leakage rate is greatly in excess of the allowable value in the technical specifications. Hence, Generic Issue C-8 was initiated in 1983 to assess (1) the causes of the MSIV leakage failures, (2) the effectiveness of the LCS and alternative mitigation paths, and (3) the need for regulatory action to limit public risk.

1.2 Regulatory and Historical Background

The requirements for control of MSIV leakage are based on 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." In particular, Criterion 54 states 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 to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits."

Criterion 55 states that:

> "Each line that is part of the reactor coolant pressure boundary and that penetrates reactor containment shall be provided with containment isolation valves. . .one automatic isolation valve inside and one automatic isolation valve outside containment. . . ."

The testing requirements for these valves are found in 10 CFR Part 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The type C test requirements in Appendix J typically result in the valves being tested every refueling outage by local pressurization with air or nitrogen at about 25 psig and a technical specification limit per valve of 11.5 standard cubic feet per hour (SCFH). The current dose assessment methodology is given in Standard Review Plan (SRP) Sections 15.6.4 and 15.6.5. The current approved licensing calculations use a fission product source term in accordance with Regulatory Guide 1.3 (Ref. 1) and do not allow credit for dose mitigation by non-safety-grade equipment in accordance with 10 CFR Part 100, Appendix A, Section III.C, which defines the safe shutdown earthquake.

In the licensing of BWR plants that had construction permits issued prior to March 1, 1970, it was assumed that the steam line and other components downstream of the MSIVs (such as the high- and low-pressure turbines and the main condenser) were undamaged when calculating the consequences of MSIV leakage in the event of a LOCA (Refs. 2 and 3). Leakage through the MSIVs was then assumed to be held up in the large volume of the condenser, with opportunity for fission product decay and plateout. As noted in Reference 2, a plateout of about 90 percent of the radioiodine was needed to get the resulting dose under Part 100 guidelines for most plants when the leakage per valve was at the typical technical specification limit of 11.5 SCFH. Since the main steam lines outside containment are usually not Seismic Category I, and the turbines and main condenser are also not Seismic Category I components, or qualified to any other safety system standards, this dependence on a non-safety-grade system to keep calculated doses below Part 100 guidelines became a staff concern after receiving reports of excessive leakage and operating malfunctions of MSIVs (Ref. 2). Hence, as discussed in Reference 2, the staff developed a position to require installation of a safety-grade MSIV leakage control system that would reduce the leakage consequences to such an extent that "dependence need not be placed on non-safety-grade piping and equipment outside the containment." Regulatory Guide 1.96 (Ref. 4), originally issued in May 1975, describes a method acceptable to the staff for meeting this requirement for all BWR plants with construction permits issued after March 1, 1970. For BWR plants with construction permits issued before March 1, 1970, the guide recommends that consideration be given to installation of an LCS if valve testing indicates recurring problems with excessive MSIV leakage. SRP Section 6.7 (Ref. 5) provides guidance for the staff review of the LCS.

As a result of this staff requirement, safety-grade MSIV leakage control systems were installed on 17 BWRs. Two types of LCS have been installed. Fourteen of these plants (8 BWR/4s, 3 BWR/5s, and 3 BWR/6s) have a negative pressure type LCS that uses blowers to produce a subatmospheric pressure in the steam lines between the MSIVs. Any leakage past the inboard MSIV is thereby collected and routed by the LCS to the safety-grade standby gas treatment system (SGTS) and discharged to the environment. Another train of this system handles the portion of the lines downstream of the outboard MSIVs. Three plants (2 BWR/4s and 1 BWR/6) have a positive pressure type LCS that provides a positive pressure in the steam lines between the MSIVs to prevent outward leakage. Twenty-one plants (1 BWR/1, 2 BWR/2s, 7 BWR/3s, 10 BWR/4s, and 1 BWR/5) do not have an LCS.

Reports of initial operating experience with leakage control systems suggested that they were prone to bothersome failures. In addition, as noted in Reference 6, staff dose calculations in 1975 had shown that the safety-grade LCS pathway gave higher offsite accident doses than a pathway involving the non-Seismic Category I main condenser. Hence, a generic issue task to investigate the desirability of the LCS was initiated in 1978 (Ref. 7). However, Generic Issue C-8, "Main Steam Line Leakage Control Systems," was characterized to be of little or no significance to plant risk (i.e., Category C).

In the early 1980s, new concerns arose because many BWR licensees had reported difficulties meeting the allowable MSIV technical specification leak rate limit during the local leak rate tests required under Appendix J. In some cases, the leak rates were much higher than allowable. A survey for the years 1979 through 1981 found that 18 of 25 operating BWRs had reported test failures. From the data, it was estimated that 58 percent of the valve tests passed the 11.5 SCFH limit, 17 percent had leakages between 11.5 and 100 SCFH, and 25 percent had leakages greater than 100 SCFH, with a mean leak rate of 1500 SCFH (Ref. 6). Since many of the latter leak rates were well above the capacity of the LCS, this mitigation path would not be available should a core damage event occur.

As the result of this experience with operating plants, Generic Issue C-8 was modified to include evaluation of problems of both excess MSIV leakage and LCS limitations and was prioritized in 1983 to be high priority in NUREG-0933 (Ref. 6). The staff calculations in NUREG-0933 included preliminary estimates of the effects of (1) reducing MSIV leakage, (2) adopting procedural changes to use an alternative non-seismic mitigation pathway instead of the LCS when available, (3) disabling the LCS at all plants, and (4) backfitting an LCS to the older plants. Of these alternatives, the reduction of MSIV leakage was clearly the major source of potential risk reduction.

Independently, the BWR Owners Group (BWROG) formed an MSIV leakage control committee to determine the causes of the high MSIV leakage rates and to develop recommendations for reducing the leakage. This BWROG committee provided three separate reports to the staff covering (1) assessment of MSIV leakage data, (2) potential operator actions to control MSIV leakage, and (3) an improved dose calculation method for assessment of consequences (Refs. 8 to 10).

The efforts of the staff and its contractor, Pacific Northwest Laboratories, through 1986 were covered in NUREG-1169 (Ref. 11). This report provided technical information on Issue C-8, but not the proposed resolution of the issue. The main elements of the effort were: (1) to evaluate the BWROG recommendations dealing with the reduction of MSIV leakage and assess the effectiveness of implementation of the recommendations by the licensees; (2) to evaluate the effectiveness of mitigation of the LCS and any alternative mitigation paths; and (3) to perform a probabilistic risk assessment of the LCS and alternative paths. In 1986, Generic Letter 86-17 (Ref. 12) was issued to all licensees and applicants of BWRs to notify them of the information available in NUREG-1169. The use of that information and the staff's intentions regarding any licensee submittal were to be formulated on a case-by-case basis.

1.3 Current Status

In NUREG-1169 it was concluded that there was a high likelihood that the BWROG had identified the key causes of high MSIV leakage and that the BWROG recommendations for corrective actions would probably solve most of the leakage problems if the recommendations were implemented. However, there were no followup leakage data available at that time to confirm this judgment. NUREG-1169 calculations indicated a number of alternative, non-safety-grade mitigation paths that gave lower offsite doses than those for the LCS. The probabilistic risk assessment (PRA) calculations, which were based on optimistic estimates of improved MSIV leakage and did not consider seismic events, showed low public risk for all pathways considered. These results and the resolution of Generic Issue C-8 were limited by the lack of confirmatory leakage data to support the BWROG recommendations on corrective actions.

In March 1988 the staff received preliminary data from a new survey by members of the BWROG (Ref. 13). The final report on these data was provided formally in August 1988 (Ref. 14) and was reviewed by PNL. The results showed a large improvement in valve leakage performance relative to the previous survey results used in the NUREG-0933 prioritization calculations. The BWROG survey included 329 test results from 24 of the 30 plants represented on the BWROG leakage control committee, or nearly two-thirds of the 38 plants now in commercial operation. It included an approximately proportional representation of valves from the three major MSIV manufacturers. Because the leakage control committee includes members from most of the plants that historically had the worst leak test problems, the data base was judged to represent the current capabilities of the whole BWR plant population. The data are interpreted to mean that the industry has identified the major causes of MSIV leakage and can successfully treat them. Therefore, credit for these new leakage tests was allowed in calculations to study the need for additional alternative actions described in this regulatory analysis.

PNL updated the analyses of NUREG–1169 to include the new BWROG leakage data and to include estimates of the effect of seismic events in the value-impact evaluation (Ref. 15).

2. OBJECTIVES

Generic Issue C–8 was initiated because of staff concerns about the high leakage rates found in routine tests of main steam isolation valves (MSIVs) and concerns about the capability of the leakage control system (LCS) that has been required to provide a safety-grade means of mitigating the effects of MSIV leakage. If the leakage of MSIVs is greatly in excess of the typical technical specification limit, the current capacity LCS may not be effective because of design limitations. In addition, calculations for other non-safety-grade mitigation paths give lower offsite doses for a given leakage than those for the LCS. The objective of this regulatory analysis is to establish the regulatory basis for the selection and implementation of alternative actions that could be taken to further reduce public risk resulting from MSIV leakage and LCS limitations.

3. ALTERNATIVES

As discussed in Section 1, the initial prioritization study of NUREG-0933 considered (1) improvements in valve leakage performance, (2) procedural changes to use an alternative non-Seismic Category I mitigation path instead of the LCS, when available, (3) disabling the LCS at all plants, and (4) backfitting an LCS to older plants as possible alternatives for reducing public risk resulting from MSIV leakage following core damage incidents. Of these options, the alternative involving improved valve leakage performance was the dominant source of potential risk reduction. Since the new improved valve leakage performance data supplied by the BWR Owners Group (BWROG) are considered to be representative of the current industry capabilities, the alternative involving reduction in MSIV leakage was not considered further and the new data were used in the risk calculations for the other alternatives.

The following alternatives were considered:

Alternative 1: No Action

This alternative would keep the current requirements and the leakage treatment practices. Plants with an LCS would remain as they are and use the LCS as the safety-grade means for treating leakage. All licensees would be expected to continue their efforts to maintain the LCS and satisfactory valve leakage performance. Consistent with the standard review plan (SRP), this alternative does not preclude a licensee from proposing some other alternative action based on plant-specific considerations.

Alternative 2: Require Addition of a Safety-Grade LCS to Plants Currently Without an LCS

This alternative considers the effects of the addition of a safety-grade LCS with standard capacity to plants that currently do not have an LCS.

Alternative 3: Require Upgrading of the Capacity of the Current LCS and the Addition of a Higher Capacity LCS to Plants Currently Without an LCS

A higher capacity LCS would be of interest to offset poor MSIV leakage performance, such as was considered in the prioritization study of NUREG–0933. The capacity of 4000 SCFH was selected to eliminate loss of availability of the LCS because of high MSIV leakage.

Alternative 4: Require Plants With an LCS to Take Them out of Service

This alternative action was considered in NUREG–0933 because of calculations indicating a higher offsite dose from the LCS (for a given MSIV leakage) than from other non-Seismic Category I paths. This alternative results in disabling of the safety-grade LCS mitigation path and dependence on one of the non-Seismic Category I mitigation paths such as considered in NUREG–1169. For this alternative, the isolated steam line path was assumed to represent the non-safety-grade path.

Alternative 5: Implement Alternative Leakage Mitigation Paths for Plants With and Without an LCS.

This alternative is similar to that considered in NUREG–0933, but involves other mitigation paths not considered at that time. For plants without an LCS, it is assumed that the isolated condenser path, one of the most effective paths found in NUREG–1169, is the preferred choice with the isolated steam line path used as a backup path for both seismic and non-seismic events. For plants with an LCS, the same non-Seismic Category I paths are used for non-seismic events. For seismic events, however, it is assumed that the safety-grade LCS path is preferred, with backup from the isolated condenser path.

4. VALUE-IMPACT METHODOLOGY AND RESULTS

This section covers the value-impact aspects of the proposed alternatives outlined in Section 3, based on the guidance given in NUREG/CR-3568, "A Handbook for Value-Impact Assessment" (Ref. 16) and in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (Ref. 17).

The value-impact attributes considered here include changes in (1) public risk, (2) occupational exposure, (3) industry implementation costs, (4) industry operating costs, and (5) NRC costs. The alternatives would not affect core damage frequency and are assumed to influence only the fission product leakage to the surroundings and the subsequent offsite conditions. Therefore, onsite costs of a core damage accident are assumed to remain unchanged. No separate estimates of offsite property damage or health costs are included since the public dose is used as the measure of all offsite radiological effects.

In this value-impact assessment, the "values" are defined as the improvements in the protection of public health and property obtained from implementation of the alternative. The "impacts" provide a measure of other consequences, mainly economic.

4.1 Methodology for Risk Calculations

As indicated above, none of the proposed alternatives should affect core damage frequency. However, the proposed alternatives would affect the release of fission products from the MSIVs to the surroundings for a given core damage incident and would change public risk. A detailed evaluation of the risk resulting from MSIV leakage for the various alternatives would involve (1) evaluation of the core damage frequency and fission product concentrations and fluid conditions in the steam lines upstream of the MSIVs used to estimate the fission product source term for the various sequences leading to core damage, (2) estimates of the probability of successful operation and decontamination factors for the various mitigation paths, (3) estimated leakage rates from the four steam lines, each with two MSIVs in series, and (4) calculations of offsite dose. For this analysis, the public risk estimates were obtained from a simplified approach based on total core damage frequencies for non-seismic and for seismic events and a single source term representing conditions upstream of the MSIVs. The effects of direct release of fission products to the surroundings and of release via the various mitigation paths were estimated and used with bounding values of the availabilities. A brief description of the public risk calculation approach is provided in this section. Additional information is given in References 11 and 15.

4.1.1 MSIV Leakage Mitigation Paths

The typical main steam system downstream of the MSIVs has four 18- to 28-inch diameter lines leading to a pressure-equalizing header. Lines lead from the header to (1) the high-pressure turbine stop and control valves, (2) the moisture separator/reheater between the high- and low-pressure turbines, and (3) the turbine bypass valves that lead to the main condenser. The pressure-equalizing header also provides steam for the feedwater pump drive turbines, steam jet air ejectors for the main condenser, offgas preheaters, clean steam reboilers, and the turbine gland seal system. Main steam line drains lead to the main condenser from several low points upstream of the turbine stop, control, and bypass valves. The main condenser air removal system consists of two trains, each with a steam jet air ejector package discharging via the offgas system to the plant stack and a mechanical vacuum pump discharging via a holdup volume to the plant stack. Vacuum breaker valves in this system permit the operator to quickly break condenser vacuum.

As discussed in Section 1, the staff required installation of a safety-grade leakage control system (LCS) on all plants with construction permits issued after March 1, 1970, because of the large number of reports of MSIV test failures and the dependence on mitigation of MSIV leakage effects by components such as the main condenser that are not seismically qualified. Brief descriptions of a typical safety-grade LCS and several alternative, non-safety-grade mitigation paths using the non-nuclear balance of plant components are provided in this section. Additional information on these paths is available in References 11 and 15.

4.1.1.1 LCS Path

Two different types of leakage control systems have been used by licensees to mitigate the effects of MSIV leakage. The first type, used in only three of seventeen plants, provides a positive back pressure of nitrogen or air to prevent leakage past the valves. Since the positive pressure type of LCS is not representative of the plants with an LCS, it was not included in the scope of this study.

The negative pressure type of LCS, which was installed on eight BWR/4s and six BWR/5s and BWR/6s, provides blowers to collect the MSIV leakage and discharge it to the surroundings via the standby gas treatment system (SGTS). Details of the LCS differ from plant to plant, but the brief description presented here is considered to represent a typical system.

A schematic diagram of the LCS path is given in Figure 4.1. Two redundant trains are provided. One train draws from the space between the inboard and outboard MSIVs. The redundant train draws from the space between the outboard MSIV and a block valve in the steam line (in BWR/6s) or the turbine stop valve. Each train exhausts in the vicinity of the SGTS in the reactor building. This discharge of the LCS would be a post-accident source of fission products in the reactor building. Although this is of concern, it was not a deciding factor for Alternatives 2 and 4 involving the addition or removal of the LCS because other significant sources of post-accident fission products exist in the reactor building, including the effects of containment leakage or rupture and operation of emergency core cooling system (ECCS) equipment (e.g., leaking pump seals).

Each train of the LCS is powered by a different diesel generator. The system is designed to be capable of performing its function following any single failure, including the failure of one MSIV to close, and to remain functional with loss of offsite power. Flow and pressure measurements and control logic are used to shut down local portions of the inboard system when the MSIV leakage is above the design capacity (typically 100 SCFH). The high-efficiency particulate air and charcoal filters in the SGTS will remove a large portion of any fission products in particulate form, elemental iodine, and organic iodines.

The LCS is designed to mitigate the consequences of MSIV leakage consistent with the containment leakage limits imposed for conditions resulting from a design basis loss-of-coolant accident (LOCA). The LCS does not have an automatic initiation feature but can be initiated manually by a key-lock switch in the control room. System initiation is prevented by interlocks until the reactor pressure is below about 20 to 35 psig. Prior to actuation of the blowers, the steam that is trapped in the main steam lines by closure of the MSIVs is bled off to the main condenser via depressurization lines. Actuation of valves in the lines to the blowers is prevented by interlocks until the steam line pressure is below about 5 psig. On the inboard system, opening of the depressurization valves for a given steam line is prevented if the associated inboard MSIV fails to close.

The operating LCS is monitored by signals indicating high reactor vessel pressure, high steam line pressure, or high MSIV leakage in order to prevent system operation under conditions that could cause system damage or be undesirable. Hence, the LCS would remain in operation unless manually terminated or system automatic isolation setpoints are exceeded. The inboard system isolates automatically on high reactor or steam line pressure or high MSIV leak rate. The outboard system isolates automatically on high reactor or steam line pressure. Restart of the system after shutdown by these signals requires additional operator action.

4.1.1.2 Alternative Mitigation Paths

Several mitigation paths for MSIV leakage that involve a number of the components in the main steam system downstream of the MSIVs were evaluated in NUREG–1169 (Ref. 11). They include (1) the steam jet air ejector (SJAE) and offgas system path, (2) the isolated steam line (ISL) path, and (3) two versions of the isolated condenser (IC) path. These paths are illustrated in Figures 4.2 to 4.4.

The isolated steam line path (see Fig. 4.2) configuration is reached when the turbine stop and control valves, the turbine bypass valves, and all branch line valves leading to the main condenser are closed. Although closed, the turbine bypass, stop, and control valves are large, fast-acting valves that are not designed to achieve tight seals. Hence, even for this path, there is some communication with the turbines and the main condenser. In NUREG–1169, it was estimated that about 60 percent of the MSIV leakage flow goes through the turbine bypass valves to the main condenser, with the remaining leakage passing through the stop and control valves to the high-pressure turbine. Since there is less holdup and plateout in the turbine before the leakage is released to the turbine building from the turbine seals, there is a higher dose than that resulting from the isolated condenser path. Calculated offsite doses for this path are roughly the same as those for the LCS path. This path corresponds to one of the steam system configurations reached on plant shutdown without operator action.

The turbine stop and control valves would be closed for typical accident scenarios. However, if either the turbine bypass valves or the main steam line drain valves are open, there is direct communication to the main condenser, and the configuration results in the isolated condenser path (see Fig. 4.3). The predicted dose consequences with the open bypass valves are somewhat lower than those with the open drain valves. This path gave one of the lowest predicted offsite doses and is one of the possible plant configurations reached on plant shutdown without operator action.

The turbine bypass valves get a closure signal at low reactor pressure. Hence, operator action would be required for opening of the turbine bypass valves. In NUREG–1169, it was estimated that, at plant shutdown without operator action, the turbine bypass valve would be open about 20 percent of the time. There was an additional small allowance for opening of the steam line drain valves on loss of offsite power. It was estimated that, for plant shutdown without operator action, the plant would be in the isolated condenser path configuration about 22 percent of the time and in the isolated steam line path configuration about 78 percent of the time. However, for some plants, the main steam line drain valves are designed for fail-safe opening, and the isolated condenser configuration would be expected most of the time.

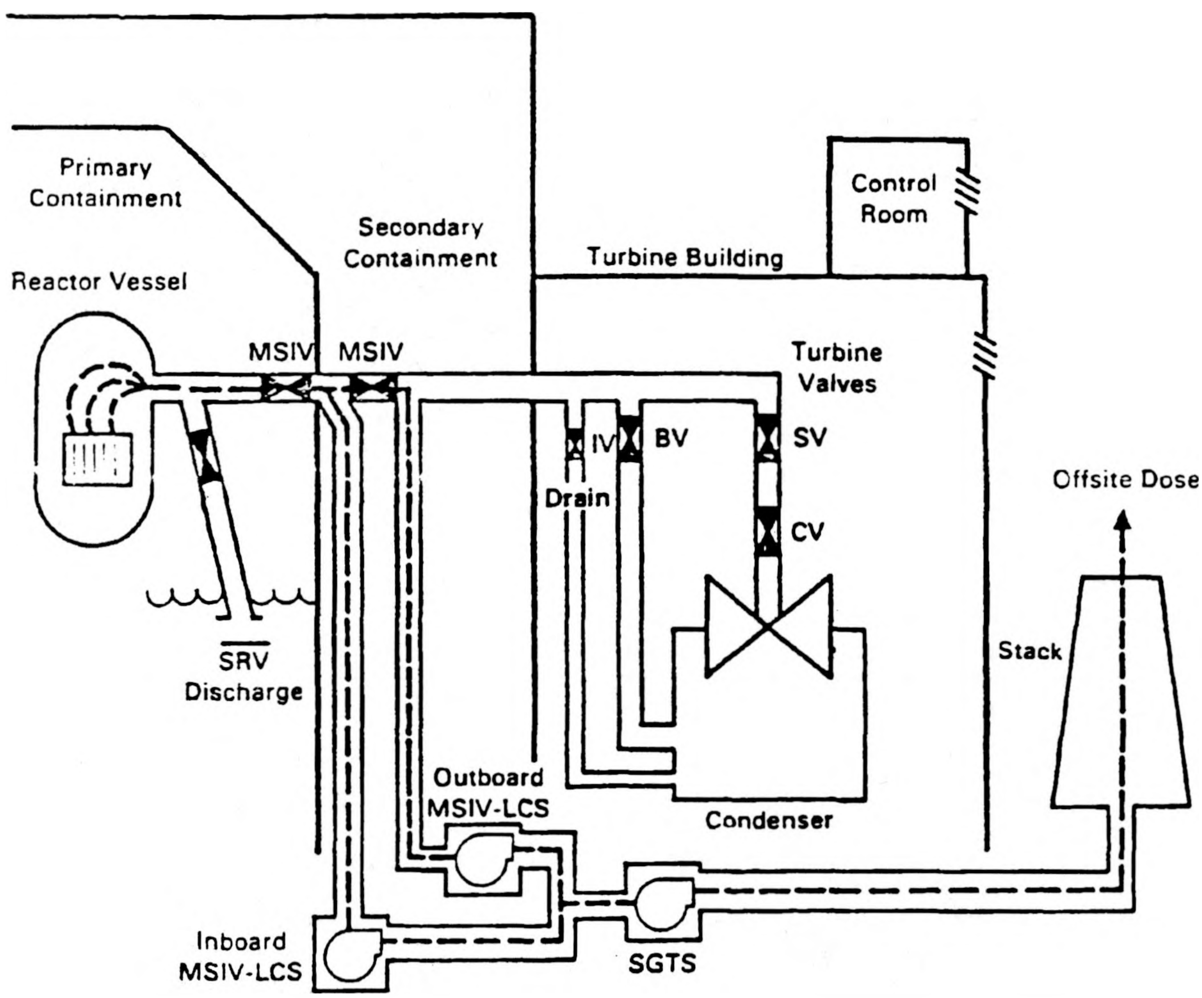

Figure 4.1 Schematic diagram of LCS path.

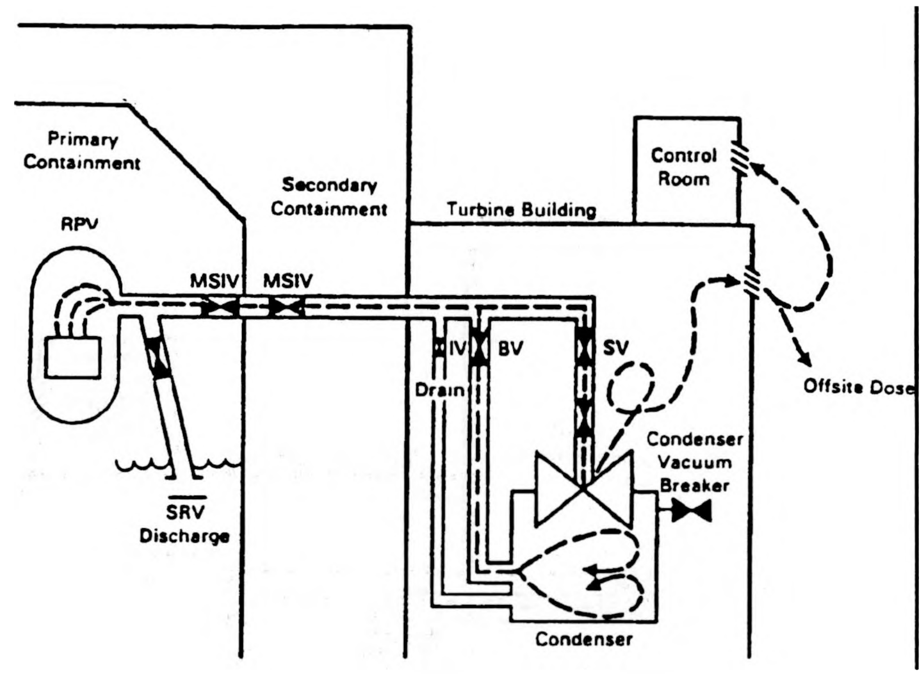

Figure 4.2 Isolated steam line path.

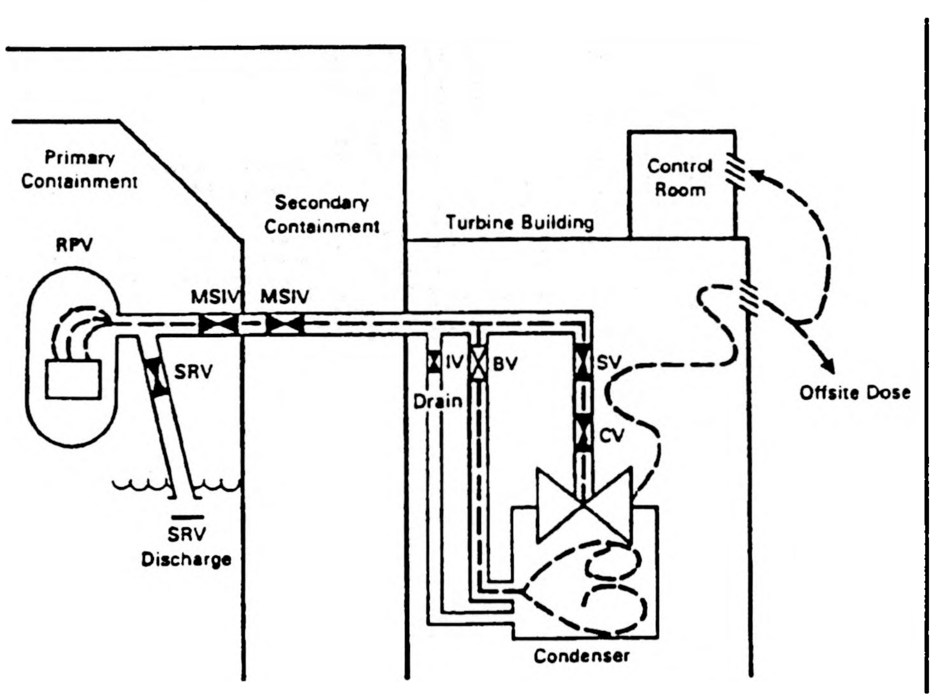

Figure 4.3 Isolated condenser path.

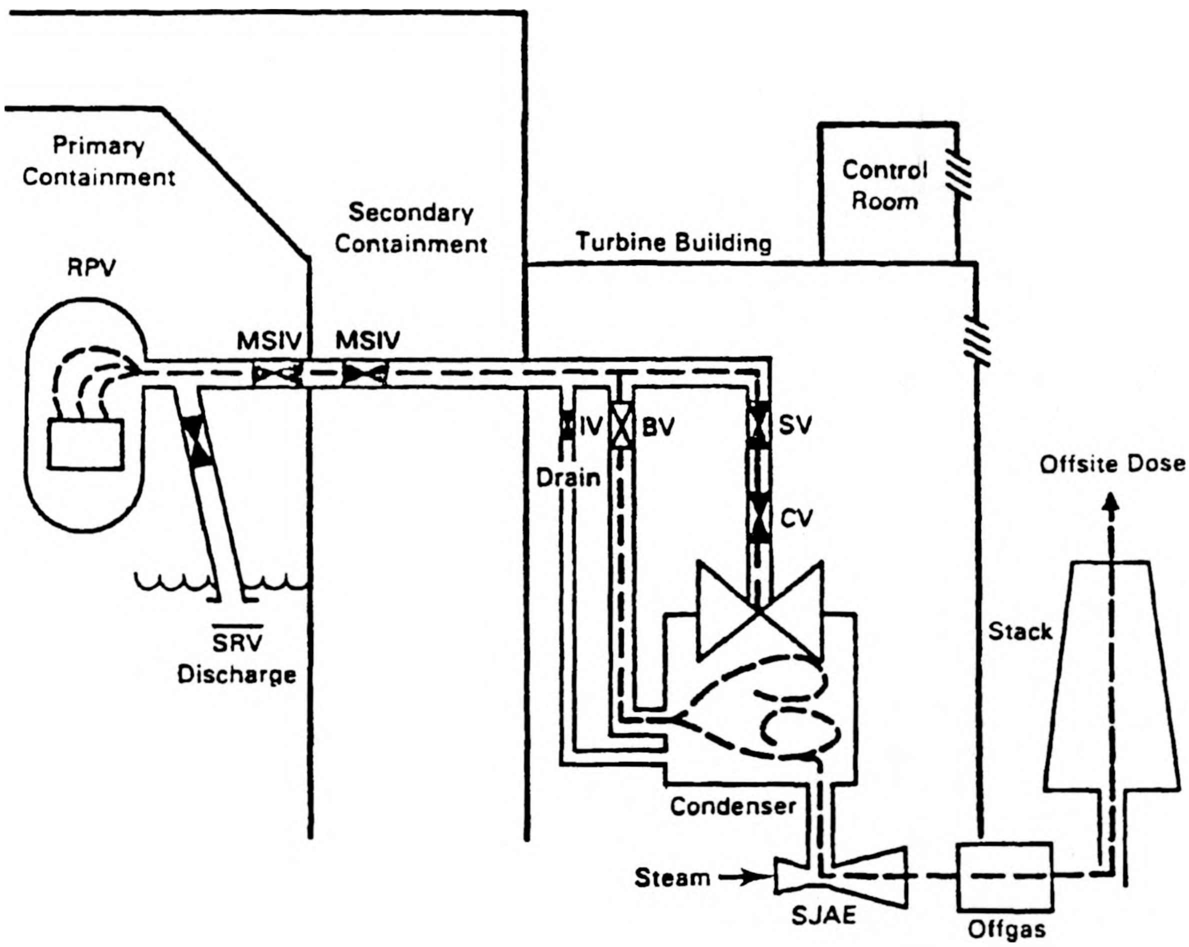

Figure 4.4 SJAE and offgas system path.

The SJAE and offgas system path (see Fig. 4.4) is very effective in the mitigation of MSIV leakage effects since it involves an elevated release point and processing by the offgas system. This path involves opening of the turbine bypass valves or steam line drain valves to the main condenser and use of the SJAEs to exhaust from the condenser to the offgas system. Since the steam supply for the SJAEs is lost with closure of the MSIVs, this mode would not be available unless an auxiliary steam supply, designed to operate the SJAEs, is available. This non-Seismic Category I path is also subject to loss of availability because of required operator actions, loss of offsite power, or loss of function of active components such as valves.

In this regulatory analysis, operator actions to use a non-Seismic Category I path were not assumed, except for one alternative involving the deliberate use of the isolated condenser path. The isolated steam line path was assumed representative of a non-Seismic Category I path reached without operator action.

4.1.2 Dose and Risk Calculation Methodology

4.1.2.1 Core Damage Frequencies

The core damage frequencies for internal and non-seismic external events used in this analysis were the following values from NUREG-1169: (1) 2.0E-4/RY for the 20 BWRs without an LCS and the 8 BWR/4s with an LCS and (2) 3.75E-5/RY for the 6 BWR/5s and BWR/6s with an LCS. On the basis of the number of plants involved and the assumed core damage frequencies, the BWR/2s, BWR/3s, and BWR/4s would be the major contributors to the risk changes for non-seismic events for the various alternatives. However, as discussed in subsequent sections, non-seismic events were a minor contributor to the risk changes for the alternatives considered in this analysis.

There are large uncertainties in the risk values associated with seismically initiated core damage events because of uncertainties in (1) core damage frequencies and (2) the seismic response of components in the mitigation paths for MSIV leakage. The mitigation paths include the Seismic Category I LCS path and the isolated condenser and steam line paths that have components that were not designed to meet Seismic Category I criteria or any other safety standards. Seismic fragility information for key components in the LCS and other paths was not available. In addition, there was very little information available for selection of core damage frequencies for seismically initiated core damage events at representative plants.

References 18 and 19 report point estimate core damage frequencies for seismically initiated core damage events of 8.26E-5/RY for Quad Cities, a BWR/3, and 8.14E-5/RY for Cooper, a BWR/4. Since these studies were done as part of a generic evaluation of decay heat removal, the scope included only the initiating events associated with small-break LOCAs and transients. Reference 20 reports a point estimate for seismically initiated core damage events of 6.0E-7/RY for LaSalle 2, a BWR/5, and a mean value of 4/0E-6/RY for Limerick, a BWR/4.

In Reference 21, the following values of core damage frequency from seismically initiated core damage events are given: Mean, 0.77E-4/RY; 95th percentile, 2.7E-4/RY. These values were based on hazard curves from Lawrence Livermore National Laboratory (LLNL). Reference 21 also gives the following corresponding values based on hazard curves developed by the Electric Power Research Institute (EPRI): Mean, 3.1E-6/RY; 95th percentile, 1.3E-5/RY. The large differences in core damage frequency were attributed to differences in methodology and the assumptions used in the development of the hazard curves.

For this analysis, the core damage frequency for seismic events was assumed to be 0.8E-4/RY for all BWRs. A high estimate of 2.7E-4/RY was assumed.

4.1.2.2 Source Term

In the evaluation of the source term for MSIV leakage, the following interactions are of interest:

1. Existing containment isolation failures could compete with the MSIV leakage for direct release to the surroundings.

2. A large MSIV leakage might reduce the probability of containment failure.

3. The source term will vary with the core melt scenario and will be affected by plateout and entrainment of fission products in the reactor vessel and coolant system, drywell, and suppression pool.

4. A containment failure associated with a core damage event could decrease the importance of the MSIV leakage.

With respect to items 1 and 2, it was concluded in NUREG-1169 that (1) existing containment failures might be comparable in consequences to, but would not reduce appreciably, the role of MSIV leakage and (2) the MSIV leakage would not play any role in preventing containment failure by pressure relief. Items 3 and 4 are discussed below with respect to the source term used in the calculations of NUREG-1169 for the various mitigation paths and for direct release.

In NUREG-1169, the leakage rate of fission products from the MSIVs was assumed to be constant for 30 days. For the containment volume of WNP-2, a test leakage of

46 SCFH was estimated to be equivalent to a containment volumetric leakage of 0.27%/day. On this basis, an MSIV test leakage of 46 SCFH was assumed equivalent to a constant leakage of 0.27%/day of the fission products initially available for release. The initially available fission products were assumed to be 100 percent of the noble gases, 25 percent of the iodines, and 1 percent of the other fission products in the reactor core fission product inventory (Ref. 22). There are a number of uncertainties in this simplified approach involving use of a single licensing type of source term. For example, the long time interval for leakage tends to overestimate total leakage but may be offset by the use of the containment volume and a 1 percent release for fission products other than the noble gases and iodines for some scenarios. Cesium is of particular interest because nearly all the core inventory of this element could be released from the fuel during a severe accident and the long-lived isotopes, Cs-134 and Cs-137, are major contributors to population exposure and land contamination.

In the evaluation of the importance of MSIV leakage, one can divide the core damage scenarios into two broad groups: (1) core damage events with associated containment failure and (2) core damage events with no containment failure. If containment fails, the MSIV leakage would contribute a very small part of the total release of fission products to the surroundings even if there were no reduction in the calculated fission product leakage through the MSIVs. Even the uncertainties in the calculated fission product release from containment failure would dominate the effect of MSIV leakage. In addition, the containment failure is a competing path for radioactive material release and should result in a much shorter time available for MSIV leakage because depressurization of containment reduces the driving force for MSIV leakage. Since current PRAs indicate a high probability of containment failure for the Mark I design (e.g., Ref. 21), this effect could have a significant impact on the perceived importance of MSIV leakage.

The vessel pressure during postulated core melt scenarios could range from containment pressure (e.g., atmospheric pressure up to containment rupture pressure) during portions of scenarios for which the vessel is depressurized, to the setpoints of the safety-relief valves for portions of scenarios for which the vessel is not depressurized. The period of time from the start of the event to the time when reactor vessel and containment failure result in a negligible driving pressure drop for MSIV leakage range from roughly 1 hour to 1 day in typical scenarios. However, the typical times from the start of core degradation to the effective end of MSIV leakage are probably less than half a day.

In view of this large overestimate in leakage time, the NUREG-1169 calculation probably gives an overestimate of the MSIV fission product leakage consequence for events involving containment failure.

There are two classes of severe accidents without containment failure that are potentially relevant to the MSIV leakage issue. For both classes the MSIV leakage could last for an extended time period (e.g., days). The first class involves TMI-like scenarios in which the core is severely damaged but is cooled in time to prevent vessel failure. The second class involves vessel failure but core arrest outside the vessel, probably as a cooled debris bed, without containment failure. For both classes of accidents without containment failure, there are two types of sequences that are of interest: (1) transients in which most of the fission products are released to the suppression pool and (2) LOCAs in which the release from the vessel is to the drywell. The LOCA-type sequences provide a larger source term in the drywell that is potentially available for release over an extended time period. Table 4.1 shows the results of a Source Term Code Package (STCP) analysis for a LOCA-initiated core melt sequence without containment failure (Ref. 23). The results indicate that, although nearly 100 percent of the core inventory of cesium is predicted to be released to the drywell, the airborne fraction in the drywell is never higher than approximately 2 percent. By five hours, the predicted concentration is much smaller than 1 percent. Hence, the assumption of 1 percent of the cesium available for release through the MSIVs for 30 days is an overestimate.

Detailed calculations with the STCP of fission product conditions in the steam lines upstream of the MSIVs for the dominant accident sequences would provide a better treatment of the fission product source term for MSIV leakage. Since these calculations were not available, the NUREG-1169 results were used. For the low MSIV leak rates in the BWROG survey, the use of this simple model was considered adequate since an allowance for a significantly larger source term at a given MSIV leak rate would not have altered the conclusions. However, if much higher leak rates are involved (e.g., from relaxation of the technical specification leak rate limits), a reevaluation of the source term and risk calculations may be warranted.

4.1.2.3 Transport Model

In the calculations of the dose reduction for various release paths in NUREG-1169, the fission products available for leakage through the MSIVs were assumed to be distributed uniformly throughout the steam phase in the primary containment atmosphere (24,960 pounds of steam) at 8 minutes following reactor scram. All radionuclides other than the noble gases were assumed to be in a particulate form. The particle size distribution was generated using the European nuclear society version of the

Table 4.1 Fraction of radionuclide group airborne in drywell for large LOCA-initiated severe accident with no containment failure.

Time (hr)	Iodine (fraction)	Cesium (fraction)
1	1.9E–2	1.9E–2
2	1.1E–2	1.1E–2
5	1.5E–5	1.5E–5

TRAP/MELT code and a large recirculation pipe break scenario. The particle size distribution resulting from agglomeration and settling of particles for 30 minutes in the drywell was assumed to represent the particles in the calculations for the mitigation paths.

The main steam lines will initially be at the normal operating temperature (550°F). Hence, for a significant amount of time after MSIV closure, the pipe wall could be hotter than the steam entering the pipes. This temperature gradient would prevent settling of small particles onto the pipe surface. To account for this effect, no credit was given for gravitational settling of particles in initially hot parts of the main steam piping for the first 48 hours.

Following MSIV closure, these lines will begin to cool by conduction to pipe hangers and other penetrations through the insulation, convective heat transfer to the ambient air from the insulation, and cooling by lower temperature steam leaking through the MSIVs and down the lines (the initially cool sections downstream of the turbine bypass valves could be heated by this steam). Using typical dimensions and properties of the pipes, insulation, and pipe hangers, it was determined by PNL in NUREG–1169 that non-uniform cooldown of the pipe, primarily by the pipe hangers, would lead to local axial temperature gradients sufficient to produce convective cells between adjacent pipe hangers and provide convective mixing of the gases in the steam lines. The calculated convective velocities were much larger than the bulk flow velocity at the lower MSIV leakage rates. These results led to the development of a model of the piping as a series of well-mixed volumes, with provisions for particle settling and radioactive decay. The PNL digital computer programs using this model to provide input for offsite dose calculations are described in NUREG–1169.

4.1.2.4 Path Availabilities

As discussed in Section 4.1.1.1, successful operation of the LCS mitigation path involves successful operator action, successful mechanical operation of the LCS and SGTS components, availability of ac power, reactor and steam line pressures below the LCS interlock and isolation setpoints, and MSIV leakage within the design capacity of the LCS. In NUREG–1169, the availability of the LCS mitigation path was estimated to be 0.8. This estimate was based on an assumed availability of 0.9 for the SGTS, a probability of successful operator action of 0.9, and an LCS availability of 0.99 from assumed availabilities of 0.9 for each of the two redundant trains. The estimate did not include consideration of the effects of seismic events on the LCS path components or the effects of partial or complete loss of ac power or reactor and steam line pressure interlock and isolation setpoints for various core damage sequences.

The partial or complete loss of the LCS during events involving station blackout (complete loss of ac power), loss of ac power for one train of the LCS, or high reactor pressure can have a major effect on the LCS availability. Since vessel pressure is at least as high as containment pressure, the system would be unavailable during periods when the containment pressure is above the reactor pressure interlock or isolation setpoints. In addition, for portions of events involving small-break LOCAs or stuck-open relief valves and the periods up to the time of meltthrough of the bottom head for an intact reactor coolant system, the system would be unavailable even for normal containment pressure. For example, the effective time window for operation of the LCS for an ATWS event with early containment failure should be limited since the vessel pressure signals prevent LCS operation until failure of the bottom head, which in turn results in loss of the driving pressure for MSIV leakage a short time later.

Seismic fragilities of the components in the LCS path and scenario pressure histories to evaluate the effect of the interlock and isolation setpoints were not available. However, the results of recent PRAs indicate a significant effect of loss of power. In the PRA for the BWR/3 plant of Reference 18, for example, 63 percent of the core damage frequency for internal events was attributed to transients where the emergency coolant injection initially succeeded, but was lost after 4 hours of operation. It was noted that these sequences primarily involved station blackout. Hence, the LCS path would have been unsuccessful for some of these sequences. For the purpose of sensitivity calculations, the high estimate of the availability of the LCS was assumed to be 1.0 for both seismic and non-seismic core damage events. However, it is noted that the availability of the LCS path in scenarios that dominate risk is probably low because of the effects of station blackout and the reactor vessel pressure signals.

As noted in Section 4.1.1.2, the plant configuration at shutdown would be expected to correspond either to the isolated steam line path or one of the two configurations (open steam line drain valves or open turbine bypass valves) for the isolated condenser path. Hence, the probability of successful operation of one of these non-Seismic Category I paths is unity for non-seismic events that do not involve component damage. However, the key components in these paths, such as the turbines, moisture-separator/reheater, and condenser, are not designed to meet Seismic Category I criteria or any other safety standards. Hence, there is a large uncertainty in the response of these components to seismic events of sufficient magnitude to cause core damage.

Following a seismic event, the various components in a passive non-Seismic Category I path might be intact or have had a breach in a pressure boundary. Hence, a seismic event could have caused partial or complete loss of the mitigation capability of the path. In this analysis, only the following two states were considered: (1) an intact path with the same decontamination factor as used for non-seismic events or (2) a disabled path with a breach in the pressure boundary resulting in a direct ground-level release to the surroundings just downstream of the outboard MSIV.

Historical information on the response to earthquakes of piping, valves, condensers, and other components for the alternative mitigation paths indicate that they can survive substantial accelerations, some much larger than safe shutdown earthquake (SSE) accelerations, without loss of structural integrity and may often retain their functional capability (Ref. 24). However, an evaluation of the seismic response of the non-Seismic Category I paths was not made for this analysis. Hence, in view of the large uncertainties in the availability of these paths during seismic events, the calculations were made for the availability at a lower limit of 0 as discussed in Section 4.2.

4.1.2.5 MSIV Leakage Data

In 1984 the MSIV Leakage Control Committee of the BWROG provided the NRC with the results of its evaluation of the causes of high MSIV leak rates. PNL reviewed these results and concluded that (1) there was a high likelihood that the committee had identified the key causes of the excessive MSIV leakage and (2) most of the leakage problems would be resolved if the committee's recommendations for corrective action were implemented (see Appendix A to NUREG-1169). When NUREG-1169 was published in 1986, there were no data available to confirm these conclusions. However, over the period from 1984 to 1987 the MSIV Leakage Control Committee recommendations on maintenance practices, valve modifications, and leak test methods were implemented by a number of licensees. The recent BWROG survey, which was formally transmitted to the NRC in August 1988 (Ref. 14) and reviewed by PNL, is summarized in Table 4.2. It shows a large improvement in performance over the leakage estimates used in the prioritization of this issue in 1983 (Ref. 6). The earlier survey had indicated that 25 percent of the tests exceeded 100 SCFH, with a mean leak rate of 1500 SCFH. In contrast, the new survey showed that (1) over 75 percent of the valves tested met the 11.5 SCFH technical specification leak rate limit, and (2) only five of the 329 tests with specified leak rates were in excess of 100 SCFH, with none over 550 SCFH. This survey included results from 24 of the 30 plants represented by the committee, or about two-thirds of the 38 plants now in commercial operation. It included an approximately proportional representation of valves from the three main valve manufacturers. In addition, the Leakage Control Committee included representatives from most of the plants that historically had the worst leak-test failure problems.

In Reference 15 PNL noted the following concerns:

1. For valves passing the leak test, the same leakage value was often assigned to both inboard and outboard valves, indicating that the actual reading for two valves was assigned equally to each valve. This would give overestimates of leak rate.
2. Several older plants with leakage problems appear to be represented and may distort the industry average upward.
3. There appear to be several newer plants with initial higher leakages that may be due to manufacturing or installation problems. This could distort the distribution to a higher value than obtained in mature plants.
4. A number of tests results are given as "greater than" some leakage rate or as "high." The latter tests can have a significant effect on the leakage over 11.5 SCFH. Since discrete data were not provided, the data could not be fit to traditional distributions.

From its review of this survey, PNL concluded in Reference 15 that the data base conservatively represented the capabilities of the BWR plant population. The data were interpreted to mean that the industry had identified the major causes of MSIV leakage problems and could correct them.

In the processing of this BWROG data for the risk calculations in Reference 15, PNL decided to combine the data from the inboard and outboard valves into one data set since there was no apparent basis for separate treatment. In addition, it was assumed that the distribution of the 29 data points with the "greater than" or "higher" designation would follow the same distribution as the 21 points with discrete values greater than 30 SCFH. A simple random number model was used to associate probability and leak rate for a single valve, as summarized in Table 4.3. The average leak rate for this distribution is about 30 SCFH.

The MSIV configuration for BWRs involves two valves in series in each of the four parallel main steam lines. It was assumed that the valves behave independently and followed the above distribution. Two random numbers were chosen independently to represent the inboard and outboard valve leakage for a given line. The smaller of the two leakages was taken as the leakage for that line. This process was repeated for the remaining three lines and the individual line leakages were summed. This process was repeated a large number of times (N = 100,000) to generate the frequency distribution for leakage from the four lines presented in Table 4.4.

Table 4.2 MSIV performance summary.

a. 300 Tests with Specified Leak Rates

Leak Rate, SCFH	Number of Valves Tested: Inboard	Number of Valves Tested: Outboard	Total Number
0 to 11.5	127	128	255
11.5 to 20	5	8	13
20 to 30	6	4	10
30 to 40	3	1	4
40 to 50	1	6	7
50 to 60	0	1	1
60 to 70	0	0	0
70 to 80	1	1	2
80 to 90	2	0	2
90 to 100	0	1	1
100 to 110	2	0	2
110 to 500	0	2	2
500 to 550	1	0	1
>550	0	0	0
Total	148	152	300

b. 29 Tests with Unspecified Leak Rates

Leak Rate, SCFH	Number of Valves Tested: Inboard	Number of Valves Tested: Outboard	Total Number
">30"	2	2	4
">37"	0	1	1
">42.3"	4	2	6
">43"	3	4	7
"High"	7	4	11
Total	16	13	29

Table 4.3 Assignment of leak rate for single value based on random number occurrence.

Random Number	Leak Rate Interval Assumed, SCFH	Leak 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

Table 4.3 Assignment of leak rate for single valve based on random number occurrence. (cont'd)

Random Number	Leak Rate Interval Assumed, SCFH	Leak Rate Assigned, SCFH
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 1.0	100 to 1000	Linear from 100 to 1000

Table 4.4 Distribution for MSIV leakage into four 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 values of Table 4.4 give an average leak rate from four steam lines of about 30 SCFH. The results also indicate that 95 percent of the events would have a leak rate from four steam lines less than about 60 SCFH.

The above calculations did not correct for (1) the reduction in leakage for some valve leakage combinations resulting from the sharing of the available pressure drop or (2) the additional block valve provided in the lines downstream of the outboard MSIV for BWR/6s. These corrections could result in a significant decrease in line leakage from the values in Table 4.4.

For some risk calculations, the results of Table 4.4 for four steam lines were collapsed to three coarse leakage intervals in which the average leak rate for the interval was assumed to be the leak rate at the upper limit of the interval. The results, shown in Table 4.5, correspond to an average leak rate of about 86 SCFH, which exceeds the 95th percentile value of Table 4.4.

Table 4.5 Coarse group leakage into four steam lines.

Leakage Interval, SCFH	Mean Leakage, SCFH	Probability
0 to 46	46	0.91117
46 to 400	400	0.08637
400 to 4000	4000	0.00246

4.1.2.6 Risk Calculations

In NUREG–1169, the outputs from the transport calculations for the various mitigation paths were used in the digital computer code CRAC2 (Ref. 25) to calculate the offsite doses for a range of MSIV leakage rates. Most of the site and reactor characteristics needed for the CRAC2 calculation were based on WNP–2 (Ref. 26). The core inventory used with CRAC2 gives the radionuclide activity for an 1120 MWe PWR. This was modified to represent WNP–2 in the present calculations by ratioing the power levels, with no correction for reactor type. The CRAC2 method for estimating the downwind dispersion of radioactivity uses the probabilistic meteorological bin sampling method and requires a data file containing a 1-year collection of hourly sequential meteorological observations. The data for WNP–2 for the year 1975, the year with the most complete available data, were used. However, a population density of 340 persons per square mile out to 50 miles was selected. This value was used in the prioritization study of Issue C–8 in Reference 6 and represents the average population density for reactor sites in the U.S.

The results of these dose calculations for the various mitigation paths is presented in Table 4.6 for three values of the total leakage rate for the four steam lines. The three leakage values in Table 4.6 are the upper limits of the three coarse leakage intervals for total leakage from the four steam lines developed in Section 4.1.2.5. The upper limit of 400 SCFH for the second interval was chosen since the LCS is assumed useful as a mitigation path only if the total leakage from all steam lines is less than this value. The table includes the CRAC2 results for a direct ground-level release to the atmosphere. These direct release results were used in the calculation of offsite dose consequences for any non-Seismic Category I path when no credit was taken for the path operability during a seismic event. The doses due to a direct release were obtained from a CRAC2 calculation for a leakage of 46 SCFH and the assumption that the dose was proportional to leak rate.

Table 4.6 Coarse group dose consequences.

Coarse Group Number	Leakage Interval for Four Steam Lines, SCFH	Probability	Average Group Leakage, SCFH	Path	Person-rem per Event
1	0 to 46	0.91117	46	LCS**	1.1E2
2	46 to 400	0.08637	400		9.7E2
3	400 to 4000	0.00246	4000		––
1	0 to 46	0.91117	46	UGLCS**	1.1E2
2	46 to 400	0.08637	400		9.7E2
3	400 to 4000	0.00246	4000		9.7E3*
1	0 to 46	0.91117	46	ISL**	3.5E1
2	46 to 400	0.08637	400		1.3E3
3	400 to 4000	0.00246	4000		2.7E4
1	0 to 46	0.91117	46	IC**	6.4E0
2	46 to 400	0.08637	400		5.6E1
3	400 to 4000	0.00246	4000		4.8E2
1	0 to 46	0.91117	46	DR**	1.2E4
2	46 to 400	0.08637	400		1.04E5
3	400 to 4000	0.00246	4000		1.04E6

* From value for 400 SCFH assuming same dose per unit SCFH.

** UGLCS = High-capacity LCS
DR = Direct ground-level release
LCS = Standard capacity LCS
IC = Isolated condenser
ISL = Isolated steam line

4.2 Value-Impact of Alternatives

4.2.1 Alternative Resolution No. 1 – No Action

This alternative keeps the current requirements, systems, and leakage treatment practices. It is assumed that plants currently with an LCS would remain as they are and use the LCS as the safety-grade means for treating leakage. All licensees would be expected to continue their efforts to maintain the LCS and satisfactory valve leakage performance. This alternative does not preclude a licensee from proposing an alternative action based on plant-specific considerations.

The results of the recent BWROG survey of MSIV leak tests give much lower average leakage rates than those derived for use in the prioritization of this issue in 1983. The data available at that time were used to generate the following coarse frequency distribution for leakage from a single MSIV:

Mean Leak Rate, SCFH	Probability
11.5	0.58
55	0.17
1500	0.25

This table gives an average leak rate of 390 SCFH, which is dominated by the contribution of 375 SCFH from the group with the mean leakage of 1500 SCFH. In NUREG-0933 (Ref. 6), this information was used to obtain an average leak rate of about 90 SCFH from a single steam line with two of these valves in series. The staff concerns in 1983 were the result of the reported high frequency of these high leak rates which (1) could bypass containment, (2) correspond to containment leakages well above the typical allowable value in the technical specifications (e.g., about 1%/day), and (3) were above the capacity of the LCS. MSIV leak rates of 1500 SCFH and 360 SCFH (the average leak rate from the four steam lines) correspond to containment leak rates of about 9 and 2%/day, respectively.

In contrast, the recent BWROG data were used to obtain an average leak rate of about 30 SCFH for a single valve and an average leakage of about 7 SCFH per steam line. This represents a reduction in average leak rate by a factor of about 13. The conservative selection of the coarse group mean leak rates for the dose calculations of Table 4.6 corresponds to an average leak rate per line of about 22 SCFH and a reduction by a factor of about 4. As discussed in Section 4.1.2.5, the new BWROG data are considered to be representative of the current capability of the BWR plant population.

If the turbine stop and control valves, turbine bypass valves, and the main steam line drain valves are closed, the plant configuration corresponds to the isolated steam line path. The decontamination factor for this path is the lowest of the non-Seismic Category I paths considered in NUREG-1169 and is roughly equal to that for the LCS path. With open turbine bypass valves or with open steam line drain valves, there is better communication between the MSIVs and the main condenser and a large increase in the mitigation. This path is called the isolated condenser path. Hence, if there were no operator action to select some other mitigation path such as the LCS or SJAE and offgas system paths, a plant with or without the LCS should end up in a configuration with a decontamination factor represented approximately by either the isolated steam line path or the isolated condenser path.

Provided there is no loss of structural integrity that could result in an assumed direct release of the MSIV leakage to the surroundings, the decontamination factors for any of the above paths result in a low risk to the public for the average MSIV leak rates representative of the current capability of the BWR plant population. This can be illustrated by considering the BWR plants in a particular group, using the total core damage frequency, and calculating the public risk due to MSIV leakage for each mitigation path, with an assumed probability of successful operation of 1.0 for the given path. For the 20 plants currently without an LCS, the average remaining life was assumed to be 25 years/plant and the total core damage frequency was assumed to be 2.0E-4/RY for non-seismic events and 0.8E-4/RY for seismic events. The values of dose per event from Table 4.6 for the three coarse groups representing MSIV leakage were used to obtain the risk estimates of Table 4.7.

Table 4.7 Risk estimates for selected mitigation paths.

Reactor Group	Event	Core Damage Frequency	Mitigation Path	Person-rem/ RY	Person-rem for Reactor Group
20 Older Plants w/o LCS	Non-Seismic	2E-4/RY	LCS	0.042	20.8
			ISL	0.043	21.3
			IC	0.002	1.2
	Seismic	8E-5/RY	LCS	0.017	8.3
			ISL	0.017	8.5
			IC	0.001	0.5
			DR	1.8	900.0

The following conclusions can be drawn from the results of this table:

1. The decontamination factor for the isolated condenser path is much larger than that for the LCS or

isolated steam line paths. However, for the average MSIV leakages considered, the dose consequences for the latter paths are so low that the additional risk reduction from use of the isolated condenser path would be small.

2. The isolated steam line path or isolated condenser path configurations cover the possible backup paths to the LCS path. Hence, for non-seismic events a large change in LCS availability results in a small change in public risk, since credit can be taken for these non-Seismic Category I paths. Also, for non-seismic events, public risk resulting from MSIV leakage should be small for plants with or without an LCS, if current industry capability for controlling MSIV leakage, as indicated by the BWROG data, is maintained.

3. During a seismic event, loss of structural integrity resulting in a breach in the pressure boundary of a key component (e.g., main condenser) in a non-Seismic Category I path might occur and result in a direct ground-level release to the surroundings. This effect, which is illustrated in the table by the entries for direct ground-level release, indicates that the core damage frequencies for seismic events and the assumptions about the seismic response of the paths would have the major effect on consequences. However, the relatively small public risk values even for the assumed direct ground-level release, suggest only small potential benefits from additional actions to reduce public risk due to MSIV leakage for the small average leakages indicated by the BWROG data.

4.2.2 Alternative Resolution No. 2 – Require Addition of an LCS with Standard Capacity to All Plants Currently Without an LCS

This alternative affects only the 21 plants currently without an LCS. The core damage frequencies for all plants in this group were assumed to be those for the older BWR/2s, BWR/3s, and BWR/4s. In the calculation of the current public risk for this group, it is assumed that the isolated steam line (ISL) path is the representative non-Seismic Category I path.

For a plant with an LCS, it is assumed that the LCS path is the preferred path for both seismic and non-seismic events. If the LCS is not available, the backup path is assumed to be the ISL path. To provide an overestimate of the reduction in public risk resulting from this alternative, the calculations are for a higher capacity LCS with no reduction in availability at the higher MSIV leak rates.

For non-seismic events, the availability of the representative non-Seismic Category I path would be 1.0, as discussed in Section 4.1. Hence, since either the LCS or the representative non-Seismic Category I path would be available, the risk for non-seismic events would be small, as discussed in Section 4.2.1. The seismic events would then dominate the change in public risk for this alternative unless the availability of the non-Seismic Category I paths during a seismic event is also close to 1.0. For this calculation, the direct release contributions for the seismic events dominate and the reduction in public risk per reactor year for this alternative is approximated by:

$$\text{DELTARISK/RY} = \text{CDFS*DOSEDR*(1-AVNSP)*AVLCS}$$

Here CDFS is the frequency for seismically induced core damage events in events per reactor year, DOSEDR is the public dose in person-rems per event for a direct ground-level release and is assumed to be proportional to the average leak rate from the four steam lines. AVNSP is the availability of the representative non-Seismic Category I path, and AVLCS is the availability of the LCS path. The total public risk reduction for the BWRs affected by this alternative is given by:

$$\text{PUBLIC RISK REDUCTION} = \text{NPLANT*REMY*DELTARISK/RY}$$

where NPLANT, the number of plants involved, is 21 and REMY, the average remaining years per plant, is assumed to be 25.

A high estimate of the public risk reduction for this plant group would be obtained by using high estimates for CDFS, AVLCS, and DOSEDR and a low estimate for AVNSP. Using the 95th percentile value of 60 SCFH for the MSIV leakage, 2.7E–4/RY for CDFS, an upper limiting value of 1.0 for AVLCS, and a lower limiting value of 0 for AVNSP, the public risk reduction per reactor year is 4.3 person-rems/RY, and the total public risk reduction is about 2200 person-rems. Calculations, including the loss of availability of the standard capacity LCS at MSIV leak rates above 400 SCFH, would give about a 15 percent lower total public risk reduction for this alternative.

This alternative involves a one-time increase in occupational exposure during installation of the LCS and the continuing occupational exposure during routine maintenance and surveillance of the LCS over the remaining reactor years. The staff cost estimate for installation of the LCS (Ref. 27) gave an estimated 6155 person-hours of plant labor at 2 millirems/hr for installation of the LCS at these older plants. The routine maintenance and surveillance is assumed to require 100 person-hours/RY at 2.5 millirems/hr over an average remaining life of 25 years for these older plants. Hence, the net change in occupational exposure resulting from this alternative is:

(21 plants) (6155 hrs/plant) (2 millirems/hr) + (21 plants) (25 yrs/plant) (100 hrs/RY) (2.5 millirems/hr) = 260 + 130 = 390 person-rems.

The industry implementation costs for this alternative are based on a staff estimate (Ref. 27) of the installation cost for an LCS with a standard capacity. The estimate included hardware and equipment costs, installation labor, engineering and quality assurance, and health physics, including anticontamination clothing. The estimate was obtained from the Phase VII Update (1986) BWR Supplement of the Energy Economic Data Base Program and modified by the staff FORECAST model. The estimated cost is:

Equipment, Hardware, and Material	\$166,500
Installation Labor Costs	\$384,000
Engineering and Quality Assurance	\$128,000
Health Physics	\$ 36,000
Total	\$714,500

It was assumed that this installation could be accommodated within a normally scheduled plant outage and that there would be no outage costs. For 21 plants, the total installation cost is \$15.0 million. It was assumed that 4 person-weeks per plant would be required to modify the technical specifications and prepare approved operating procedures. At \$50/person-hour, this industry cost would be:

(21 plants) (4 person-wks/plant) (40 hrs/wk)(\$50/hr) = \$0.17 million.

Hence, the total industry implementation cost for installation of the LCS and documentation is \$15.2 million.

The net change in industry operating costs for this alternative is based on an assumed increase of 100 person-hours per reactor year for routine maintenance and surveillance operations for the LCS. With an assumed labor cost of \$40 per hour and an average remaining plant life of 25 years, the industry operating costs, discounted to 1989 dollars with a 10 percent discount rate is:

(21 plants) (100 person-hours/RY) (\$40/hr) (9.077) = \$0.8 million.

The NRC costs to follow the installation of the LCS is estimated to be 2 person-weeks per plant. Since changes to the technical specifications are needed, an NRC cost of \$10,400 per plant is estimated based on an assumption of 4 person-weeks of technical time, 2 person-weeks of management and legal time, and \$800 for *Federal Register* notices. For the 21 plants, the estimated NRC costs are:

(21 plants) (2 person-wks) (40 hrs/wk) (\$40/hr) + (21 plants) (\$10,400/plant) = \$286,000.

The above contributions give the best estimate value for the net change in total industry and NRC costs for this alternative as an increase of \$16.3 million. This increase is dominated by the installation cost for the LCS. The major uncertainty for installation costs is considered to be the possible additional costs if the installation could not be completed within the normally scheduled plant outage. Since this effect was not included, correction for this uncertainty would lead to a higher cost. Assuming a 30 percent allowance for uncertainty in cost, the low estimate for total industry and NRC cost is \$11.4 million.

From the above results, the low estimate of the value-impact ratio (net change in total industry and NRC costs divided by public risk reduction) is \$11.4 million/2200 person-rems or \$5200/person-rem. Hence, an additional factor of 5 increase in the dose for a given MSIV leak rate to allow for uncertainties in the source term is available. If the best estimate values for CDFS (0.8E–4) and MSIV leak rate from four steam lines (30 SCFH) were used with the same limiting values for AVLCS (1.0) and AVNSP (0), the value-impact ratio would increase by about a factor of 7 to \$35,000/person-rem.

If the value term is changed from the net change in public risk to the net change in the sum of public risk plus occupational exposure, the low estimate of the value-impact ratio would become \$11.8 million/(2200–390) person-rem, or about \$6500/person-rem. For this case, use of the best estimate value for CDFS and MSIV leak rate with the limiting values for AVLCS and AVNSP would result in a small increase in the sum of the public risk and occupational exposures for this alternative.

4.2.3 Alternative Resolution No. 3 – Require Upgrading of the Capacity of a Currently Installed LCS and Addition of a Higher Capacity LCS to Plants Currently Without an LCS

A higher capacity LCS would be a possible alternative if current MSIV leakage tests still indicated a high frequency of high leakage rates such as used in NUREG–0933. In the evaluation of this alternative, the capacity of the higher capacity LCS was arbitrarily taken as 4000 SCFH to correspond to the upper limit of 1000 SCFH per valve used in the treatment of the BWROG test data. This permitted elimination of loss of availability of the system because of high MSIV leakage. The higher capacity system was assumed to give the same offsite dose per unit leakage from the steam lines as the standard capacity LCS. Hence, the differences in dose consequences between this system and the standard LCS result from leakages from the four steam lines in excess of 400 SCFH. For the low MSIV leak rates of the BWROG survey, there is a small change in LCS availability with increase in LCS capacity. The reduction in public risk resulting from installation of a more costly higher capacity LCS is bounded by the results of Alternative 2. In addition, upgrading of the standard capacity LCS would result in a decrease in public risk that is less than about 10 percent of the risk reduction of Alternative 2 and, hence, should be less cost effective than Alternative 2.

4.2.4 Alternative Resolution No. 4 – Disable the LCS at All Plants

This alternative involves only the 14 plants currently with a negative pressure LCS. With removal of the LCS, it is assumed that the preferred paths change from the LCS to the isolated steam line path used in Alternative 1. The magnitude of the net change in public risk for this alternative differs from that of Alternative 2 because it involves a smaller number of plants with an assumed average remaining life of 34 years/plant. The high estimate of the change in public risk for this alternative is an increase of about 2000 person-rems. If the best estimate values for CDFS (0.8E-4/RY) and MSIV leak rate into four steam lines (30 SCFH) were used with the same values for AVLCS (1.0) and AVNSP (0), the above value would decrease by about a factor of 7.

This alternative results in a one-time increase in occupational exposure to take the LCS out of service at 14 plants and a decrease in occupational exposure because the routine maintenance and surveillance operations for this system would no longer be needed. It is estimated that 100 person-hours at 2.5 millirems/hr would be needed to disable the LCS. It is assumed that 200 person-hours per reactor year at 2.5 millirems/hr are needed for maintenance and surveillance operations for a typical current LCS. The resulting net change in occupational exposure is a decrease of 235 person-rems.

The industry costs to take the LCS out of service at the 14 plants involves a one-time labor cost to disable the LCS plus the cost to revise the technical specifications and operating procedures. It is estimated that 100 person-hours of plant labor at $40/hr would be needed to disable the LCS and 160 person-hours at $50/hr would be needed for the documentation changes. The resulting net change in industry implementation costs for this alternative would be an increase of $0.17 million. In addition, after the disabling of the LCS, the routine maintenance and surveillance costs for the 14 plants with an assumed average remaining life of 34 years would no longer be needed. Assuming 300 person-hours per year for this operation, and discounting to 1989 dollars with a 10 percent discount rate, this would result in a net decrease in operating costs of $1.61 million. The total net change in industry costs is a decrease of $1.44 million.

The NRC costs to implement this alternative would be staff time to formulate and approve the directive to licensees; revise regulatory guides, standard technical specifications, and standard review plans; and obtain management and legal review. This is estimated to require 40 person-weeks. In addition, about 1 person-week and $10,400 per plant would be required for review of changes to plant technical specifications. The total costs would be:

$$\begin{gathered}(1600 \text{ person-hrs})(\$40/\text{hr}) + (14 \text{ plants})(\$10{,}400/\text{plant}) \\ + (14 \text{ plants})(40 \text{ person-hrs/plant})(\$40/\text{hr}) \\ = \$0.23 \text{ million.}\end{gathered}$$

This one-time cost increase would be partially offset by a small cost saving resulting from elimination of any NRC followup costs related to the LCS, giving a net cost of $0.21 million.

The net change in industry and NRC costs resulting from implementation of this alternative is a decrease of $1.23 million for the 14 plants in this group.

4.2.5 Alternative Resolution No. 5 – Implement Alternative Mitigation Paths for Plants With and Without an LCS

As shown in NUREG-1169, the two versions of the isolated condenser path (open steam line drain valves or open turbine bypass valves), the steam jet air ejector path, and the mechanical vacuum pump (or turbine gland exhauster) path had approximately the same calculated decontamination factors that were larger than those for the LCS and isolated steam line paths. Of these, the two versions of the isolated condenser path and the isolated steam line paths involve the configurations considered to represent the endpoint configurations that would be reached without operator action or the need for power.

In this alternative, it is assumed that, for a non-seismic event, the operator selects one of these more effective paths as the preferred path and that the isolated steam line path represents the backup path. For seismic events, it is assumed that the LCS would be selected as the preferred path, if available, and that the isolated condenser path represents the backup path.

The results show that the differences in public risk for the various mitigation paths are very small. The net change in public risk for this alternative would be less than about 20 person-rems. This alternative would involve additional industry and NRC costs associated with documentation changes. These costs were not estimated because of the small predicted risk reduction.

5. DECISION RATIONALE

This generic issue deals with staff concerns about public risk because of the excessive number of failures of the MSIVs for BWRs to meet the allowable technical specification leak rate limit and the limitations of the LCS required at later plants to mitigate the consequences of MSIV leakage. The issue was initiated in 1983 to evaluate (1) the causes of the MSIV leakage failures, (2) the effectiveness of the LCS and alternative mitigation paths, and (3) the need for additional regulatory action to reduce public risk. Independently, the BWROG formed the MSIV Leakage Control Committee to determine the causes of the high MSIV leakage rates and develop recommendations for reducing the leakage. The BWROG committee provided the staff with reports dealing with this assessment of leakage, potential operator actions to control MSIV leakage, and improved calculation methods to assess the dose consequences of MSIV leakage.

The efforts of the staff and its contractor, PNL, (through 1986) were covered in NUREG–1169. Calculations in this report indicated that several non-Seismic Category I mitigation paths have larger decontamination factors than the negative pressure LCSs currently installed at BWRs. It was concluded that the BWROG recommendations for improving the MSIV leakage performance could solve most of the leakage problems, if implemented. However, there were no data available at that time to confirm this judgment. In August 1988 the results of a new BWROG survey of MSIV leak tests from 1984 through 1987 were formally transmitted to the staff. This survey, which shows a significant improvement in MSIV leakage performance, was reviewed by PNL and considered to represent the current capabilities of the BWR plant population. PNL updated the risk calculations of NUREG–1169 to include the new BWROG data and estimates of the effects of seismic events in the value-impact evaluation done in support of this regulatory analysis (Ref. 15).

On the basis of the results of this value-impact analysis and the PNL evaluation of the new BWROG data, it is concluded that no backfit requirement to limit public risk associated with MSIV leakage or the LCS is warranted and that Alternative 1 should be adopted. This alternative maintains the current requirements, systems, and leakage treatment practices. This conclusion is based on the following considerations:

1. The new BWROG survey involved 24 of 30 plants represented on the BWROG Leakage Control Committee, or about two-thirds of the 38 BWRs now in commercial operation. Of the 329 tests involving representation from the three major valve manufacturers, 77 percent met the 11.5 SCFH specification and only one test reported a leak rate in excess of 500 SCFH. The average leak rate was estimated to be about 30 SCFH for a single valve and 7 SCFH for a single steam line with two valves in series. In contrast, the MSIV data available in 1983 indicated that only 58 percent of the valves passed the leak test, about 25 percent had leak rates with a mean value of 1500 SCFH, and the average leak rate for a single valve was about 390 SCFH. This new data base for MSIV leakage was judged to represent the current capability of the BWR population and was interpreted to mean that the industry has identified the major causes of MSIV leakage and can correct them.

   It is noted that an increase in MSIV leakages for the BWR industry to levels that are significantly outside the range of data assumed in this analysis would be cause for reevaluation of these conclusions. MSIV leak rates (although not a generic problem at this time) should not be allowed to return to the high observed leak rates noted prior to 1983. Hence, continued staff surveillance of industry experience with MSIVs and continued efforts by industry to maintain satisfactory MSIV performance are warranted. Although increases in MSIV leak rate translate into increased public risk, considerations of modest increases in allowable leakage may be justifiable on a case-by-case basis.

2. The cost-benefit analyses of Alternatives 2 and 3, dealing with (1) the addition of standard capacity LCSs to plants currently without an LCS and (2) adding higher capacity LCSs and upgrading the capacity of the LCSs currently installed at BWRs show that neither alternative is cost-effective.

3. Alternative 4 involved removal of the LCS from the 14 plants that currently have a negative pressure LCS. This resulted in an estimated net savings of $1.23 million for the 14 plants in this group and a high estimate of an increase in public risk of 2000 person-rems. The net change in the sum of public risk and occupational exposure resulting from this alternative is estimated to range from an increase of about 1800 person-rems to a decrease of less than 200 person-rems. In view of the uncertainties in the estimates and the small net changes involved, it is concluded that there is insufficient basis for a generic requirement to remove the LCS from operation, although plant-specific requests to remove the LCS may be justifiable. Such requests for removal of the LCS may require an exemption from 10 CFR Part 100 concerning credit for non-safety-grade paths after removal of the LCS.

4. Alternative 5, which involved the use of other non-Seismic Category I paths with larger decontamination factors resulted in a very small reduction in public risk. Hence, a generic requirement to implement this alternative is not warranted. A generic letter that referred to the possible benefit of these paths

was sent to all licensees, and the paths have been evaluated by the BWROG.

6. IMPLEMENTATION

No regulatory action is necessary for resolution of this issue. This regulatory analysis and the PNL report in support of this regulatory analysis will be made publicly available as part of the normal distribution.

REFERENCES

1. U.S. Nuclear Regulatory Commission (USNRC), Regulatory Guide 1.3, "Assumptions used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Revision 2, June 1974.

2. USNRC Memorandum from J.M. Hendrie to J.F. O'Leary et al., "BWR Main Steam Line Isolation Valve Seal Requirement," February 27, 1973.

3. USNRC Memorandum from H.R. Denton to R. Boyd, "Response to Technical Assistance Request No. BWR 2–4, James A. Fitzpatrick Nuclear Power Station (TAR–310)," January 30, 1973.

4. USNRC, Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," Revision 1, June 1976.

5. Standard Review Plan Section 6.7, "Main Steam Isolation Valve Leakage Control System (BWR)," NUREG–0800, July 1981.

6. R. Emrit et al., "A Prioritization of Generic Safety Issues," USNRC Report NUREG–0933, December 1983.

7. USNRC, "Generic Task Problem Description—Category B, C, and D Tasks," NUREG–0471, September 1978.

8. Section 2.1, "Main Steam Isolation Valve Historical Data Evaluation and Assessment," of Appendix A to NUREG–1169, August 1986.

9. "Potential Operator Actions to Control MSIV Leakage," General Electric Co., NEDO–30324, September 1985.

10. H. Careway et al., "A Technique for Evaluation of BWR MSIV Leakage Contributions to Radiological Dose Rate Calculations," General Electric Co., NEDO–30259, September 1985.

11. J. N. Ridgely and M. L. Wohl, "Technical Findings Related to Generic Issue C–8: Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods," USNRC Report NUREG–1169, August 1986.

12. USNRC Memorandum from R. Bernero to all BWR Project Managers, "Generic Letter 86–17 Related to Publication of NUREG–1169 Concerning Generic Issue C–8, MSIV Leakage and LCS Failure," October 17, 1986.

13. USNRC Memorandum from C. Graves to K. Kniel, "BWROG Data for MSIV Leakage," May 31, 1988.

14. Letter with attachment from D. N. Grace, BWROG Chairman, to C. Graves, USNRC, "BWR Owners' Group MSIV Leakage Closure Committee Recent MSIV Leakage History and Cost Benefit Data, Revision 2," August 15, 1988.

15. J. Jamison et al., "Value-Impact Analysis of Regulatory Options for Resolution of Generic Issue C–8: MSIV Leakage and LCS Failure," Pacific Northwest Laboratory, NUREG/CR–5397, PNL–6931, May 1990.

16. S.W. Heaberlin et al., "A Handbook for Value-Impact Assessment," Pacific Northwest Laboratory, NUREG/CR–3568, PNL–4646, December 1983.

17. USNRC, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," NUREG/BR–0058, Revision 1, May 1984.

18. S.W. Hatch et al., "Shutdown Decay Heat Removal Analysis of a Generic Electric BWR3/Mark I—Case Study," Sandia National Laboratories, NUREG/CR–4448, SAND85–2373, March 1987.

19. S.W. Hatch et al., "Shutdown Decay Heat Removal Analysis of a General Electric BWR4/Mark I—Case Study," Sandia National Laboratories, NUREG/CR–4767, SAND86–2419, July 1987.

20. P.G. Prassinos, "Evaluation of External Hazards to Nuclear Power Plants in the United States: Seismic Hazard," Lawrence Livermore National Laboratory, NUREG/CR–5042, Supplement 1, UCID–21223, April 1988.

21. USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG–1150, Vols. 1 and 2, Second Draft for Peer Review, June 1989.

22. J.J. DiNunno et al., "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, TID-14844, March 1962.

23. R. S. Denning et al., "Source Term Calculations for Assessing Radiation Dose to Equipment," Battelle Memorial Institute, Columbus Laboratories, NUREG/CR-4949, BMI-2153, July 1989.

24. American Society of Civil Engineers, "The Effects of Earthquakes on Power and Industrial Facilities and Implications for Nuclear Power Plant Design," 1986.

25. L.T. Ritchie et al., "Calculations of Reactor Accident Consequences, Version 2, CRAC2 Computer Code: User's Guide," Sandia Laboratories, NUREG/CR-2326, SAND81-1994, April 1983.

26. WPPSS Nuclear Project No. 2, Final Safety Analysis Report, Washington Public Power Supply System, Richland, Washington.

27. Memorandum from S. Feld to C. Graves, "Cost Estimates for Regulatory Analysis—Generic Issue C-8: MSIV Leakage and LCS Failure," September 16, 1988.

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-1372

2. TITLE AND SUBTITLE

Regulatory Analysis for the Resolution of Generic Issue C-8, "Main Steam Isolation Valve Leakage and LCS Failure"

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	1990

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Charles C. Graves

6. TYPE OF REPORT

Regulatory

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION – NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

9. SPONSORING ORGANIZATION – NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

Generic Issue C-8 deals with staff concerns about public risk because of the incidence of leak test failures reported for main steam isolation valves (MSIVs) at boiling water reactors and the limitations of the leakage control systems (LCS) for mitigating the consequences of leakage from these valves. If the MSIV leakage is greatly in excess of the allowable value in the technical specifications, the LCS would be unavailable because of design limitations.

The issue was initiated in 1983 to assess (1) the causes of MSIV leakage failures, (2) the effectiveness of the LCS and alternative mitigation paths, and (3) the need for additional regulatory action to reduce public risk. This report presents the regulatory analysis for Generic Issue C-8 and concludes that no new regulatory requirements are warranted.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Generic Safety Issue C-8
Main Steam Isolation Valve
Leakage Control System
Boiling Water Reactors

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page) Unclassified

(This Report) Unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)