U.S. NUCLEAR REGULATORY COMMISSION

May 1975

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

REGULATORY GUIDE 1.96

DESIGN OF MAIN STEAM ISOLATION VALVE LEAKAGE CONTROL SYSTEMS FOR BOILING WATER REACTOR NUCLEAR POWER PLANTS

A. INTRODUCTION

General Design Criterion 54, "Piping Systems Penetrating Containment," of Appendix A, "General Design Criteria," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires, in part, that piping systems penetrating primary containment be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating these piping systems. This guide describes a basis acceptable to the NRC staff for implementing General Design Criterion 54 with regard to the design of a leakage control system for the main steam isolation valves of boiling water reactor nuclear power plants to ensure that total site radiological effects do not exceed guidelines of 10 CFR Part 100, "Reactor Site Criteria," in the event of a postulated design-basis loss-of-coolant accident (LOCA).

B. DISCUSSION

Direct cycle boiling water nuclear power plants supply steam directly from the reactor vessel to the turbine via main steam lines. The main steam lines installed on current BWR plants are provided with dual quick-closing isolation valves. These valves function to isolate the reactor system in the event of a break in a steam line outside the primary containment, a design-basis LOCA, or other events requiring containment isolation. In the case of a steam line break, the isolation valves would terminate the blowdown of reactor coolant in sufficient time to prevent an uncontrolled release of radioactivity from the reactor vessel to the environment. In the case of a LOCA, the valves would isolate the reactor from the environment and prevent the direct release of fission products from the containment.

The valves are part of the reactor coolant pressure boundary. As such they are Quality Group A components and their integrity must be maintained by strict inservice inspection and testing requirements. However, operating experience has indicated that degradation has occasionally occurred in the leak tightness of main steam isolation valves such that the specified low leakage requirements have not always been maintained continuously.

The staff has considered the need to provide additional features to ensure the low-leakage characteristics of the main steam isolation valves in the event of a postulated design-basis loss-of-coolant accident.[1] The use of a leakage control system would reduce direct untreated leakage from the isolation valves when isolation of the primary system and the containment is required.

The results of staff analyses have indicated that calculated doses resulting from the maximum leakage allowed under the technical specification for the main steam isolation valves in postulated design-basis LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines,[2] provided the main steam system from the isolation valves up to and including the turbine

[1] In its letters on the construction permit reviews of the Duane Arnold and Shoreham plants (December 18, 1969) and the James A. FitzPatrick plant (January 27, 1970), the Advisory Committee on Reactor Safeguards (ACRS) noted that such features to control main steam isolation valve leakage should be considered.

[2] Part 100 guidelines, as used in this guide, refer to the radiation dose limits used in determining the boundaries of the exclusion area and the low population zone pursuant to 10 CFR Part 100.

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condenser remains intact. However, with the standard conservative assumption used by the staff in considering the offsite consequences of postulated design-basis LOCA, e.g., loss of leak tightness beyond the turbine stop valve, uncontrolled valve leakage at or above current typical technical specification limits of 11.5 standard cubic feet per hour per valve at typical calculated containment pressures could result in calculated doses in excess of Part 100 guidelines.

The position of the staff with respect to the seismic design classification of the steam system does not require Seismic Category I design requirements for the turbine stop and control valves, steam line piping beyond the stop valve, the turbine, the turbine condenser, or connecting piping of less than 2½ inches in diameter. However, there is a need for design improvements to provide appropriate safety margins for the large numbers of plants now planned. The staff believes that, unless systems can be relied upon to remain intact and capable of providing significant dose reduction factors in postulated accident conditions, a leakage control system for main steam isolation valves should be provided for new boiling water reactor plants[3] to supplement the isolation function of the main steam isolation valves and reduce uncontrolled or untreated leakage from the steam line valves.

It has been proposed that dose reduction factors due to the transport delay time of the containment atmosphere in passing through the main steam lines within containment or through the main steam lines from the isolation valves to the turbine stop valves should be included in staff calculations of postulated accident effects. Analyses by some applicants, based on assumptions different from those used by the staff, have indicated that long transport delays might occur. On that basis, it has been argued that a leakage control system is not necessary to reduce potential leakage from the steam systems of boiling water reactor plants. The staff has considered these analyses and has concluded that, although they are useful in making so-called "realistic" or "best-estimated" dose calculations and hence in showing margins that might exist above the limit-type calculations of the staff, a more positive method of reducing the radiological effects of potential leakage of the main steam system isolation valves should be provided. The staff also has concluded that some limited credit for transport delay effects is appropriate in determining the design basis for such leakage control systems.

Staff analyses of the contribution of main steam isolation valve leakage to total calculated offsite doses in postulated design-basis loss-of-coolant accidents made with conservative allowances for transport delay effects show that the two-hour site boundary dose is not affected by the subject leakage. The long-term dose in the low population zone, however, is affected for uncontrolled isolation valve leakage rates typical of current technical specification values. Thus, the staff has concluded that a fully automatic quick-acting leakage control system is not required to meet the system objectives. A manually initiated leakage control system capable of being actuated within about 20 minutes of an accident requiring use of the system would be acceptable.

It should be noted that any leakage from the stem packing of the outboard isolation valve would contribute to the two-hour dose, since in most designs such leakage would escape to the turbine building and the environment via the steam tunnel. Reduction and control of steam packing leakage or other direct leakage to the steam tunnel from the outboard isolation valve should be a design objective of the leakage control system or of other systems provided for this purpose.

C. REGULATORY POSITION

The isolation function of the main steam isolation valves in boiling water reactor plants should be supplemented by a leakage control system (LCS). An acceptable approach for such a leakage control system is provided by the following design basis:

1. The leakage control system and any necessary subsystems, including the source of any sealing fluid if a fluid seal type of system is used, should be designed in accordance with Seismic Category I and Quality Group B requirements, with the exception of any portion of LCS piping that connects to main steam system piping between inner and outer containment isolation valves of the main steam system for either single- or dual-barrier containment structures. Such piping, up to and including the first isolation valve in the LCS piping, should be designed in accordance with Seismic Category I and Quality Group A requirements supplemented by Appendix A of this guide.

2. The LCS (and any necessary subsystems) should be capable of performing its safety function, when necessary, considering effects resulting from a LOCA, including (a) missiles that may result from equipment failures, (b) dynamic effects associated with pipe whip and jet forces, and (c) normal operating and accident-caused local environmental conditions consistent with the design-basis event. Further, any portion of the LCS which is Quality Group A and is located outside the primary containment structure should be protected from missiles, pipe whip, and jet force effects originating outside containment such that containment integrity is maintained.

3. The LCS should be capable of performing its safety function following a LOCA and assumed single active failure (including failure of any one of the main steam isolation valves to close).

4. The LCS should be designed so that effects resulting from a single active component failure of the

[3]The staff defines "new" boiling water plants to be those plants utilizing the General Electric Company's BWR 6/Mark III design or subsequent BWR designs.

leakage control system will not affect the integrity or operability of the main steam lines or main steam isolation valves.

5. The LCS should be capable of performing its safety function following a loss of all offsite power coincident with a postulated design-basis LOCA.

6. The LCS should be designed with sufficient capacity and capability to control leakage from the main steam lines for as long as postulated accident conditions require containment integrity to be maintained.

7. The LCS may be manually or automatically actuated and should be designed to permit actuation within about 20 minutes after a postulated design-basis LOCA. This time period is considered to be consistent with loading requirements of the emergency electrical buses and with reasonable times for operator action.

8. Instrumentation and circuits necessary for the functioning of the LCS should be designed in accordance with standards applicable to an engineered safety feature.

9. The LCS controls should include interlocks to prevent inadvertent operation of the LCS. In particular, interlocks should be provided to prevent damage to the LCS or possibly to the main steam system due to inadvertent opening of any LCS isolation valves whenever the pressure in the connecting main steam piping exceeds LCS design pressure. All such controls and interlocks should be activated from appropriately designed safety systems or circuits.

10. The plant should be designed to permit testing of the operability of controls and actuating devices of the LCS during power operation to the extent practical and testing of the complete functioning of the system during plant shutdowns.

11. The LCS should be designed so that any effects resulting from use of a fluid sealing medium, such as thermal stresses, pressures associated with flashing, and thermal deformations under the loading conditions associated with the activated system, will not affect the structural integrity or operability of the main steam lines or main steam isolation valves, and that any deformation of isolation valve internals will not induce leakage of the main steam line isolation valve beyond the capacity or capability of the LCS.

12. Equipment should be provided, as part of the LCS or other systems, to prevent or control valve stem packing leakage or other direct leakage from main steam line isolation valves outside containment. If such equipment is not part of the LCS, it should meet the same design standards as the LCS.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for utilizing this regulatory guide.

This guide reflects current regulatory practice. Therefore, except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used in evaluating construction permits and operating license applications docketed after publication of this guide. Although this guide may recommend backfitting in certain cases that were docketed prior to publication of this guide, as described below, it does not require it. Such requirements will be formulated on an individual basis pursuant to 10 CFR §50.109.

1. For boiling water reactor plants for which construction permits were issued prior to March 1, 1970, as listed in Table 1, applicants and licensees should continue the established inservice inspection programs to ensure that isolation valves are maintained in such a manner that leakage is within Technical Specification limits. If the valve inspections show recurring problems with excessive leakage, the staff recommends that consideration be given to installation of a supplementary leakage control system.

2. For boiling water reactor plants of designs preceding the BWR 6/Mark III design, and for which construction permits have been issued after March 1, 1970, as listed in Table 1, the staff recommends that applicants and licensees install a supplemental leakage control system. Leakage control systems for these plants should be designed in accordance with the basis stated in the regulatory position of this guide to the extent practical considering the stage of plant design and construction. The system provided for each plant and the schedule for installation will be reviewed on an individual-case basis.

3. For boiling water reactor plants of the BWR 6/Mark III design (or subsequent BWR design), including and subsequent to the first BWR 6/Mark III (i.e., the Grand Gulf project), the staff recommends that applicants and licensees install a supplemental leakage control system to ensure the isolation function of the main steam isolation valves. Leakage control systems for the plants should be designed in accordance with the basis stated in the regulatory position of this guide.

TABLE 1

LIST OF BWR PLANTS

SECTION D.1 PLANTS	DATE OF ACRS CP REPORT	CP ISSUED
Dresden 1	7/55	5/56
Big Rock Point	3/60	5/60
Humboldt Bay	6/60	11/60
Lacrosse	12/62	3/63
Oyster Creek	8/64	12/64
Nine Mile 1	10/64	4/65
Dresden 2	11/65	1/66
Millstone 1	3/66	5/66
Dresden 3	8/66	10/66
Quad Cities 1, 2	12/66	2/67
Browns Ferry 1, 2	3/67	5/67
Monticello	4/67	6/67
Vermont Yankee	6/67	12/67
Peach Bottom 2, 3	10/67	1/68
Cooper	3/68	6/68
Browns Ferry 3	5/68	7/68
Pilgrim 1	4/68	8/68
Hatch 1	5/69	9/69
Brunswick 1, 2	10/69	2/70
SECTION D.2 PLANTS		
FitzPatrick	1/70	5/70
Duane Arnold	12/69 & 2/11/70	6/70
Fermi 2	2/71	9/72
Zimmer	9/71	10/72
Hatch 2	11/71	12/72
Hanford 2	10/72	3/73
Shoreham	12/69 & 2/70	4/73
LaSalle 1, 2	12/71	9/73
Susquehanna 1, 2	4/72	11/73
Hope Creek 1, 2 (ex. Newbold Island)	8/71 & 2/74	Pending
Limerick 1, 2	8/71	6/74
Bailly	10/71	5/74
Nine Mile 2	7/73	6/74

APPENDIX A

SUPPLEMENTAL DESIGN FEATURES FOR QUALITY GROUP A PORTION OF LCS PIPING

This appendix provides supplemental design features for any portion of piping for a leakage control system that connects to steam system piping between inner and outer containment isolation valves of the main steam system for either single- or dual-barrier containment structures. Such piping, up to and including the first isolation valve in the LCS piping, should be constructed to meet the requirements of the ASME Code in Section III, Subarticle NE-1120, supplemented by the following:

1. The following design stress and fatigue limits should not be exceeded:

a. The maximum stress range should not exceed $2.4S_m$.

b. The maximum stress range between any two load sets (including the zero load set) should be calculated by Eq. (10) in Par. NB-3653, ASME Code, Section III, for upset plant conditions and an operating basis earthquake transient.

If the calculated maximum stress range of Eq. (10) exceeds $2.4S_m$ but is not greater than $3S_m$, the cumulative usage factor should be less than 0.1.

If the calculated maximum stress range of Eq. (10) exceeds $3S_m$, the stress ranges calculated by both Eq. (12) and Eq. (13) in Par. NB-3653 should not exceed $2.4S_m$ and the cumulative usage factor should be less than 0.1.

2. Welded attachments, for pipe supports or other purposes, to this portion of piping should be avoided.

3. The number of circumferential and longitudinal welds in the piping should be minimized.

4. The portion of piping extending to the first shutoff valve should be as short as practical.

5. The design of piping restraints should not require welding directly to the outer surface of the piping.

6. The design of this portion of the leakage control system should permit the conduct of inservice examinations required by the rules of Section XI of the ASME Boiler and Pressure Vessel Code, and the extent of examinations during each inspection interval should provide 100 percent volumetric examination of the piping welds within this portion of piping.