Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

August 2, 2004

L-2004-178
10 CFR 50.54(q)

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Proposed Change to Emergency Plan
Table 3-1: Classification of Emergencies

Pursuant to 10 CFR 50.54(q), Florida Power & Light Company (FPL) requests approval prior to implementation, of a change to the St. Lucie Plant Radiological Emergency Plan.

FPL requests NRC approval for changing the Initiating Condition (IC) for an (Notification of) Unusual Event (UE) due to reactor coolant system (RCS) leakage to resolve an unintended consequence of the 1996 IC change. FPL seeks NRC approval to: (1) reestablish the link to the RCS Technical Specification applicability; and (2) add the condition "ability to isolate" when determining the occurrence of RCS leakage. The current IC and the proposed IC remain consistent with the NUREG 0654 scheme of classification. The ICs for Alert and above remain unchanged by this request.

The proposed change has been reviewed by the Facility Review Group (FRG) on July 29, 2004. On receipt of approval from the NRC of the proposed change, FPL will revise the St. Lucie Plant Emergency Plan and Emergency Plan Implementing Procedures to implement the revised UE declaration threshold. FPL will obtain the concurrence of state and local governments prior to implementing this change.

FPL discussed the attached proposed IC change and the schedule with the NRC Project Manager NRC Staff on July 23, 2004. FPL requests the NRC to complete the review by November 1, 2004, to support training for implementation prior to the fall 2004 Unit 2 refueling outage (SL2-15) which is currently scheduled to start in late November.

Please contact George Madden at 772-467-7155 if there are any questions about this submittal.

Very truly yours,

William Jefferson, Jr.
Vice President
St. Lucie Plant

WJ/GRM

Attachments (3)

A045

Attachment 1

FPL requests NRC approval for changing the Initiating Condition (IC) for an (Notification of) Unusual Event due to Reactor Coolant System (RCS) leakage to resolve an unintended consequence of the 1996 IC change. FPL seeks NRC approval to: (1) reestablish the link to RCS Technical Specification applicability; and (2) add the condition "ability to isolate" when determining the occurrence of RCS leakage. The current IC and the proposed IC remain consistent with the NUREG 0654 scheme of classification. The contents of this submittal package are as follows:

A. Background
B. Comparison of current IC/EAL to proposed IC/EAL
C. Basis and justification for the change
D. State/local government review/concurrence
E. Supporting References

A. BACKGROUND

In 1996, St. Lucie Plant requested and received NRC approval for changes submitted for the Initiating Condition (IC) for an (Notification of) Unusual Event (TAC Nos. M96274 and M96275). That change allowed an Unusual Event (UE), due to reactor coolant system (RCS) leakage, to be determined solely on the basis of the quantity of the leak and not whether the leakage was identified or unidentified. As a consequence of that change, the RCS Emergency Action Levels (EAL) were no longer tied to the leak rates defined in the Unit 1 or Unit 2 Technical Specifications and therefore, not dependent on the mode relationship within those specifications. The change inappropriately broadened the applicability of the EALs to all modes defined in Technical Specifications. The EALs used at St. Lucie Plant are based on NUREG 0654. The IC change approved in 1996 was an acceptable alternative to the NUREG 0654 IC previously in place. This proposed change seeks to revise the original acceptable alternative by re-establishing a link to the Technical Specifications and therefore mode dependence as originally defined in the NUREG 0654 RCS leakage IC.

B. COMPARISON OF CURRENT IC/EAL TO PROPOSED IC/EAL

Refer to Attachment 2.

C. BASIS AND JUSTIFICATION FOR THE CHANGE

St. Lucie currently uses the NUREG 0654 scheme of classification. The NRC safety evaluation dated October 17, 1996 (TAC Nos. M96274 and M96275) allowed the RCS leakage Unusual Event to be characterized as >10 GPM in an effort to avoid the time consuming evolution of doing a mass balance prior to determining if an emergency condition exists. When that EAL was changed, the reference to Technical Specification was removed, feeling that we were not using the Technical Specification leak rate as defined in the specification. That deletion resulted in no longer bounding RCS leakage to conditions described in the specification, specifically it now includes all reactor operating modes, an approach inconsistent with NUREG-0654 guidance and more in keeping with the NEI/NUMARC scheme of emergency action levels (CU-1 for Mode 5).

St. Lucie has historically focused on the reactor coolant system as a liquid in lieu of the physical system, making it difficult to determine if a challenge to the RCS barrier had occurred. As such, it was necessary to establish a clear definition of the physical system called the reactor coolant system. To that end, a review was performed of the Updated Final Safety Analysis Report (UFSAR), NUREG-1432, The Standard Technical Specifications for Combustion Engineering Plants, the Code of Federal Regulations, NEI 99-01 Revision 4, Methodology for Developing Emergency Action Levels, and St. Lucie's current Technical Specifications and their bases.

From UFSAR Section 5.1, the Reactor Coolant System (RCS) circulates water in a closed cycle,

> *"to remove heat from the reactor core and transfers it to a secondary (steam generating) system. ...The major components of the system are the reactor vessel; two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps; a pressurizer connected to one of the reactor vessel outlet pipes; and associated piping. All components are located inside containment."*

NUREG-1432, The Standard Technical Specifications for Combustion Engineering Plants (STS) defines the reactor coolant system as components that contain or transport the coolant to or from the reactor core and the Technical Specification covers Modes 1 through 4 specifically.

With regards to RCS leakage, the STS states:

> *"In Modes 1, 2, 3, and 4, the potential for reactor coolant pressure boundary leakage is greatest when the RCS is pressurized. In Modes 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in low stresses and reduced potentials for leakage."*

The Code of Federal Regulations, Title 10 section 50.2, includes the following definition of Reactor Coolant Pressure Boundary:

> *"Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:*
>
> *(1) Part of the reactor coolant system, or*
> *(2) Connected to the reactor coolant system, up to and including any and all of the following:*

> *(i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,*
> *(ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,*
> *(iii) The reactor coolant system safety and relief valves."*

In NEI 99-01 Revision 4, Methodology for Developing Emergency Action Levels, the definition for the reactor coolant system is as follows:

> *"The RCS Barrier includes the RCS primary side and its connections up to and including the Pressurizer safety and relief valves, and other connections up to and including the primary isolation valves."*

The St. Lucie Technical Specification for RCS leakage (T.S. 3.4.6.2) contains the following Limiting Condition for Operation (LCO) for Modes 1, 2, 3, and 4 (Unit 2 specific is in brackets):

> *Reactor Coolant System leakage shall be limited to:*
>
> *a. No PRESSURE BOUNDARY LEAKAGE,*
> *b. 1 GPM UNIDENTIFIED LEAKAGE,*
> *c. 1 GPM total primary-to-secondary leakage through steam generators,*
> *d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and*
> *e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.*
> *[1 gpm leakage (except as noted in Table 3.4-1) at a Reactor Coolant System pressure of 2235 + 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.]*

As a result of the above review, St. Lucie has incorporated the following definition for RCS for Emergency Plan use:

> *"RCS includes any component (pipe, vessel, valve, etc.) which is used to contain or transport the reactor coolant to or from the reactor core. This definition includes any component beyond the RCS pressure boundary, which remains open to the RCS."*

According to the STS definition of RCS, systems connected to the reactor coolant pressure boundary (RCPB) should be considered as an extension of the RCS when not isolated from the RCS. During normal plant operations, the RCS extension would include the chemical and volume control system (CVCS) letdown and charging lines since they are open to the RCS. During other conditions, systems that may be included under the definition provided above include shutdown cooling (SDC)/low pressure safety injection (LPSI), high pressure safety injection (HPSI), containment spray (CS), RCS sample, and some portions of the waste management system. Leakage from one of these systems, when it remains open to the RCS, would be considered reactor coolant leakage, since it would make it impossible to determine RCS barrier leakage, as such, it would allow an uncontrollable reduction of the RCS inventory being used for core cooling, thus potentially compromising plant safety.

The concept of isolating secondary systems from the physical RCS is evident in the initial NRC guidance provided to the industry in NUREG-0818, Emergency Action Levels for Light Water Reactors. In the category of RCS Technical Specification leakage for the Notification of Unusual Event EAL (page 24), the NRC found the draft EAL submitted by the V. C. Summer Plant acceptable for meeting the NUREG-0654 EAL (allowing for the timeframe provided in Technical Specification for returning the leak into conformance with the specification). This concept is also in place elsewhere in the industry.

FPL is in agreement that isolating interfacing systems is a primary means of determining if there is a true challenge to the reactor coolant system barrier. Furthermore, allowing plant operations a reasonable amount of time to isolate those interfacing systems is prudent. The initial steps operators take in response to excess RCS leakage would be considered an appropriate timeframe.

The proposed change involves declaration of an Unusual Event due to RCS leakage. According to the class description, Notification of Unusual Event (NUREG-0654) is indicative of "*unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.*" The NRC has approved the definition of RCS in NEI 99-01 Revision 4 which states "*The RCS Barrier includes the RCS primary side and its connections up to and including the Pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.*" Additionally, The NRC has stated in Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, that "*the safety significance of leaks from the reactor coolant pressure boundary (RCPB) can vary widely depending on the source of the leak as well as the leakage rate and duration.*"

The two factors of concern regarding RCS leakage, as stipulated in the Regulatory Guide, are: (1) leakage rate, which is addressed by the established threshold of 10 gpm in the current EAL of the IC; and (2) the duration of the leakage, which is addressed by requiring isolation of the leak in the proposed change to the IC. An additional concern stated in both the Regulatory Guide and the NUREG is the location of the leak. RCS leakage detection capabilities within containment are unchanged. Detection of RCS leakage outside of containment is what is impacted by this proposed change. The proposed IC for Unusual Event due to RCS leakage addresses both the duration of leakage and RCS leakage outside of containment/RCPB. A leak of RCS in excess of 10 gpm must be readily isolable within the bounds of initial operator action or an emergency is declared. A leak in an interfacing system that exceeds 10 gpm and is non-isolable requires an emergency declaration. The Technical Specification allows for isolating the high pressure portion of the system from the low pressure systems in an effort to understand the leakage. The expectation for isolation/termination of the leak is promptly, with promptly being within the bounds of initial operator actions in off-normal operating procedures or emergency operating procedures.

In accordance with 10 CFR 50.54 (q), a "*licensee may make changes to these plans [Emergency Plan] without Commission approval only if the changes do not decrease the effectiveness of the plans and the plans, as changed, continue to meet the standards of 10 CFR 50.47(b) and the requirements of appendix E to this part.*" The proposed change seeks to revise the original acceptable alternative by re-establishing a link to the Technical Specifications and therefore mode dependence as originally defined in the NUREG-0654 RCS leakage IC. Implementation of this change would eliminate the basis for entry into the Emergency Plan for conditions (i.e. Mode 5 and 6) that, prior to this change, would have implemented the Emergency Plan. No RCS leakage Technical Specification is provided for the St. Lucie Plant for Modes 5 or 6. In fact, NUREG-0654 does not include a low mode emergency action levels for RCS leakage. The current EAL scheme for St. Lucie addresses the low mode conditions through the IC for the inability to maintain cold shutdown and the loss of sub-cool margin. Technical Specification RCS leakage in excess of 10 gpm that is isolable does not "*indicate a potential degradation of the level of safety of the plant.*" FPL feels that this change to the Emergency Plan enhances the program in that it no longer unnecessarily focuses offsite emergency management attention on a non-emergency condition.

The proposed revision of this IC remains in agreement with the NUREG-0654 scheme of emergency classification and the class description for Unusual Event and continues to meet 10 CFR 50.47 (b) and Appendix E. The revised IC will also continue to provide a logical transition to the IC for Alert within the classification table event/category, "Abnormal Primary Leak Rate." If a RCS leak were in excess of 50 gpm and unisolable, then conditions would require declaration of Alert.

The proposed change provides an alternate to the existing IC/EAL that is more in line with NUREG-0654 but less restrictive than the current IC/EAL. Therefore, FPL seeks NRC approval prior to implementation of this change to the Emergency Plan.

D. STATE/LOCAL GOVERNMENT REVIEW/CONCURRENCE

FPL will obtain the concurrence of state and local governments prior to implementing this change.

E. SUPPORTING REFERENCES

The following documents, in total or in part, have been included as Attachment 3 to this submittal:

E.1 Safety Evaluation of Proposed Emergency Action Level Revision for St. Lucie Plant, Units 1 and 2 (TAC NOS. M96274 and M96275).

E.2 St. Lucie Plant Updated Final Safety Analysis Report, Unit 1 – Chapter 5, Reactor Coolant System and Unit 2 – Section 5.0, Reactor Coolant System and Connected Systems.

E.3 NUREG-1432, Vol.1, Rev. 3.0, Standard Technical Specifications Combustion Engineering Plants, June 2004.

E.4 NRC Regulations, Title 10, Code of Federal Regulations Part 50.2, Definitions.

E.5 NEI 99-01 Rev. 4 (NUMARC/NESP-007), Methodology for Development of Emergency Action Levels, January 2003.

E.6 St. Lucie Plant Technical Specifications Unit 1 – 3.4.6.2, Reactor Coolant System and Unit 2 – 3.4.6.2, Reactor Coolant System.

E.7 NUREG-0818, Emergency Action Levels for Light Water Reactors, October 1981.

E.8 NUREG-0654, FEMA-REP-1, Rev.1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, November 1980.

E.9 Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.

Attachment 2
Comparison of Current IC/EAL to Proposed IC/EAL

NUREG 0654	Current IC/EAL	Proposed IC/EAL
5. Exceeding either primary/secondary leak rate technical specification or primary system leak rate technical specification. 6. Failure of a safety or relief valve in a safety related system to close following reduction of applicable pressure.	Reactor Coolant System (RCS) Leakage 1. RCS leakage GREATER THAN 10 gpm as indicated by: A. Control Room observation OR B. Inventory balance calculation OR C. Field observation OR D. Emergency Coordinator Judgment OR 2. Indication of leaking RCS safety or relief valve which causes RCS pressure to drop below SIAS set points: - Unit 1 - 1600 psia - Unit 2 - 1736 psia	Reactor Coolant System (RCS) Leakage 1. *Unisolable* Technical Specification RCS leakage GREATER THAN 10 gpm as indicated by: ***NOTE*** • *If the leak is from an interfacing system (e.g., SDC, LPSI, CVCS, etc.) and the leak is readily isolable from the Reactor Coolant Pressure Boundary, the leak should not be considered RCS leakage.* • *To be isolable, personnel must be able to promptly close the valve(s) which isolates the leak within the context of initial operator actions.* A. Control Room Observation OR B. Inventory balance calculation OR C. Field observation OR D. Emergency Coordinator's judgment OR 2. Indication of leaking RCS safety or relief valve causes RCS pressure to drop below SIAS setpoints: - Unit 1 – 1600 psia - Unit 2 – 1736 psia

Attachment 3

Supporting References

E.1 Safety Evaluation of Proposed Emergency Action Level Revision for St. Lucie Plant, Units 1 and 2 (TAC NOS. M96274 and M96275).

E.2 St. Lucie Plant Updated Final Safety Analysis Report, Unit 1 – Chapter 5, Reactor Coolant System and Unit 2 – Section 5.0, Reactor Coolant System and Connected Systems.

E.3 NUREG-1432, Vol.1, Rev. 3.0, Standard Technical Specifications Combustion Engineering Plants, June 2004.

E.4 NRC Regulations, Title 10, Code of Federal Regulations Part 50.2, Definitions.

E.5 NEI 99-01 Rev. 4 (NUMARC/NESP-007), Methodology for Development of Emergency Action Levels, January 2003.

E.6 St. Lucie Plant Technical Specifications Unit 1 – 3.4.6.2, Reactor Coolant System and Unit 2 – 3.4.6.2, Reactor Coolant System.

E.7 NUREG-0818, Emergency Action Levels for Light Water Reactors, October 1981.

E.8 NUREG-0654, FEMA-REP-1, Rev.1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, November 1980.

E.9 Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

RECEIVED
OCT 22 1996
Nuclear Licensing

October 17, 1996

Mr. Thomas F. Plunkett
President, Nuclear Division
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408-0420

Dear Mr. Plunkett:

SUBJECT: SAFETY EVALUATION OF PROPOSED EMERGENCY ACTION LEVEL REVISION FOR ST. LUCIE PLANT, UNITS 1 AND 2 (TAC NOS. M96274 AND M96275)

By letter dated July 25, 1996, Florida Power and Light proposed a revision to Table 3-1 of the St. Lucie Plant Emergency Plan. NRC approval was requested, prior to implementation, of a change to an Emergency Action Level (EAL) regarding reactor coolant system (RCS) leakage. The proposed change revises the declaration threshold for the EAL of an Unusual Event involving RCS leakage.

The NRC staff has completed its review of the proposed change and found that the proposed EAL meets the requirements of 10 CFR 50.47(b)(4) and Appendix E to 10 CFR 50 for emergency classification and action level schemes. Our Safety Evaluation is enclosed. Section IV.B of Appendix E to 10 CFR Part 50 requires agreement by State and local government authorities to changes to the plant's EALs. In a telephone call with George Madden of your staff on October 11, 1996, Mr. Madden stated that agreement with State and local officials would be obtained prior to EAL change implementation, or the change would not be implemented. Based upon this commitment, you are hereby authorized to implement this change to the St. Lucie Plant Radiological Emergency Plan conditioned upon your obtaining the agreement of appropriate State and local officials prior to implementation thereof. This completes our action on TAC Nos. M96274 and M96275.

Sincerely,

Leonard A Wiens, Senior Project Manager
Project Directorate II-3
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Docket No. 50-335
and 50-389

Enclosure: Safety Evaluation

cc w/enclosure: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AN EMERGENCY ACTION LEVEL REVISION

FLORIDA POWER AND LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

By letter dated July 25, 1996, Florida Power and Light Company (FPL) proposed a revision to Table 3-1 of their Emergency Plan. They requested approval, prior to implementation, of a change to an Emergency Action Level (EAL) regarding reactor coolant system (RCS) leakage. The proposed EAL change revises the declaration threshold for an Unusual Event involving RCS leakage.

2.0 BACKGROUND

The proposed EAL change was reviewed against the requirements in 10 CFR 50.47(b)(4) and Appendix E to 10 CFR 50. Section 50.47(b)(4) specifies that onsite emergency plans must meet the following standard: "A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee...."

Section IV.C. of Appendix E to 10 CFR 50 specifies that, "Emergency action levels (based not only on onsite and offsite radiation monitoring information but also on readings from a number of sensors that indicate a potential emergency, such as the pressure in containment and the response of the Emergency Core Cooling System) for notification of offsite agencies shall be described.... The emergency classes defined shall include: (1) notification of unusual events, (2) alert, (3) site area emergency, and (4) general emergency."

The current EAL followed the general guidelines for EALs set forth in Appendix 1 of NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants" (November 1980). The proposed revision incorporates the enhancements and clarifications to the EALs based on the guidelines for EALs set forth in NUMARC/NESP-007, Revision 2, "Methodology for Development of Emergency Action Levels" (January 1992). The NRC endorsed the use of either NUREG-0654 or NUMARC/NESP-007 in Regulatory Guide 1.101, "Emergency Planning and Preparedness for Nuclear Power Reactors," Revision 3, August 1992. Regulatory Guide 1.101 provides acceptable methods by which licensees may meet the requirements of 10 CFR 50.47(b)(4) and Appendix E to 10 CFR 50. In Emergency Preparedness Position (EPPOS) Number 1, "Emergency Preparedness Position on Acceptable Deviations from Appendix 1 of NUREG-0654 Based Upon the Staff's Regulatory Analysis of NUMARC/NESP-007" (June 1, 1995), the NRC staff recognized that NUREG-0654-based EALs could be enhanced and clarified by application of the technical bases for NUMARC/NESP-007-based EALs. The staff relied upon the guidance in these documents as the basis for its review of the St. Lucie proposed EAL revision.

3.0 EVALUATION

The licensee followed the guidance and logic presented in NUMARC/NESP-007 for determining the level of leakage to declare an Notification of an Unusual Event (NOUE). The licensee revised the threshold for an NOUE and EAL based upon RCS leakage from two values:

(1) greater than 1 gpm for unidentified leakage and

(2) greater than 10 gpm for identified leakage;

to one threshold value:

greater than 10.gpm for all leakage.

The new proposed threshold leakage value is higher for unidentified sources than the current EAL and is the same as leaks from identified sources. The value for identified leaks are more conservative than the value shown in NUMARC guidance. However, the licensee indicated that the selected value and the establishment of a single threshold value makes the declaration of an NOUE quicker and easier. The determination of the quantity and the source of the leak will be simpler and more timely under the proposed EAL. The licensee will not have to confirm the leakage levels through mass inventory balances and the licensee will not have to spend time to differentiate between identified and unidentified leaks.

Although Regulatory Guide 1.101 admonishes against the mixing of the emergency classification guidance in NUMARC/NESP-007 with that in Appendix 1 to NUREG-0654, it is recognized that licensees who continue to utilize the example initiating conditions in Appendix 1 to NUREG-0654 as the basis for their classification scheme could benefit from the guidance in NUMARC/NESP-007. To that end, licensees could utilize the technical bases under the example EALs in NUMARC/NESP-007 to enhance and clarify some of their site-specific EALs developed from NUREG-0654. The chosen classification scheme, whether based on Appendix 1 to NUREG-0654 or NUMARC/NESP-007, must remain internally consistent.

The staff found the proposed revisions and associated justification, provided by the licensee, to be acceptable. The proposed EAL revision for the St. Lucie Plant, is consistent with the guidance provided by NUREG-0654 and allowable deviations, as discussed in EPPOS 1 in accordance with the technical bases for EALs in NUMARC/NESP-007.

4.0 CONCLUSION

As a result of our review, we have concluded that the proposed EAL meets the requirements of 10 CFR 50.47 and Appendix E to 10 CFR Part 50 for emergency classification and action level schemes. However, in addition to NRC approval of EAL changes, Section IV.B of Appendix E to 10 CFR Part 50 requires agreement by State and local government authorities to changes to the plant's EALs. Therefore, the licensee should obtain such agreement prior to implementation of the proposed EAL.

Principal Contributor: L. Cohen

Dated: October 17, 1996

CHAPTER 5

REACTOR COOLANT SYSTEM

This chapter was originally prepared to describe the reactor coolant system during the initial fuel cycle. Much of the original text is retained for historical record. However, where applicable, changes have been made to reflect the uprating of the unit to a stretch power level of 2700 Mwt. Where information associated with the higher power level is not available the existing information is identified as "pre-stretch" or "cycle 1."

5.1 SUMMARY DESCRIPTION

The function of the reactor coolant system is to remove heat from the reactor core and transfer it to the secondary (steam generating) system. In a pressurized water reactor the steam generators represent the points of separation between the reactor coolant system and the main steam system. The steam generators are vertical U-tube heat exchangers in which heat is transferred from the reactor coolant to the main steam system. Reactor coolant is separated from the boiler water by the steam generator tube sheet. The reactor coolant system is a closed system which forms a barrier to the release of radioactive materials into the containment.

Plan and elevation views of the arrangement of the reactor coolant system are shown in Figures 5.1-1 and 5.1-2, respectively. The piping and instrumentation (P&I) diagram of the reactor coolant system is shown in Figure 5.1-3. The major components of the system are the reactor vessel, two heat transfer loops, each containing one steam generator and two reactor coolant pumps, a pressurizer connected to the loop 1B reactor vessel outlet pipe; and connecting inlet and outlet, spray and surge line piping. A quench tank is provided to receive, condense, and cool steam discharges from the pressurizer safety and power operated relief valves. All components are located inside the containment, and the relationship of the equipment arrangement to the containment structure is shown in Figure 1.2-7 through 1.2-11.

Table 5.1-3 shows the principal pressures, temperatures, flow rates and coolant volumes of the reactor coolant system components under pre-stretch normal steady state, full power operating conditions by means of numbered locations (See Figure 5.1-3). Figure 5.1-3 has a detailed representation of the reactor coolant system. Instrumentation provided for operation and control of the system is described in Section 7.

System pressure is maintained by regulating the water temperature in the pressurizer where steam and water are held in thermal equilibrium. Steam is either formed by the pressurizer heaters or condensed by the pressurizer spray to limit the pressure variations caused by contraction or expansion of the reactor coolant. The pressurizer is located with its base at a higher elevation than the reactor coolant loop piping. This eliminates the need for a separate pressurizer drain, and ensures that the pressurizer is drained before maintenance operations. The average temperature of the reactor coolant varies with power level, and the fluid expands or contracts, changing the pressurizer water level.

5.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

This chapter was originally prepared to describe the reactor coolant system during the initial fuel cycle. Much of the original text is retained for historical record. However, where applicable, changes have been made to reflect the uprating of the unit to a stretch power rating of 2700 Mwt. Where information associated with the higher power level is not available the existing information is identified as Cycle 1.

5.1 SUMMARY DESCRIPTION

The reactor is a pressurized water reactor with two coolant loops. The Reactor Coolant System (RCS) circulates water in a closed cycle, to remove heat from the reactor core and transfers it to a secondary (steam generating) system. The steam generators provide the interface between the Reactor Coolant (primary) System and the Main Steam (secondary) System. The steam generators are vertical U-tube heat exchangers in which heat is transferred from the reactor coolant to the Main Steam System. Reactor coolant is prevented from mixing with the main steam by the steam generator tubes and the steam generator tube sheet. The RCS is a closed system thus forming a barrier to the release of radioactive materials.

The arrangement of the RCS is shown on Figures 5.1-1 and 5.1-2. The major components of the system are the reactor vessel; two parallel heat transfer loops, each containing one steam generator and two reactor coolant pumps; a pressurizer connected to one of the reactor vessel outlet pipes; and associated piping. All components are located inside containment.

Reactor Coolant System pressure is controlled by the pressurizer, where steam and water are maintained in thermal equilibrium. Steam is formed by energizing immersion heaters in the pressurizer, or is condensed by the pressurizer spray to limit pressure variations caused by contraction or expansion of the reactor coolant. The average temperature of the reactor coolant varies with power level and the fluid expands or contracts, changing the pressurizer water level.

The charging pumps and letdown control valves in the Chemical and Volume Control System (CVCS) are used to maintain a programmed pressurizer water level. A continuous but variable letdown purification flow is maintained to keep the RCS chemistry within prescribed limits. Two charging nozzles and a letdown nozzle are provided on the reactor coolant piping for this operation. The charging flow is also used to alter the boron concentration or correct the chemical content of the reactor coolant.

Other reactor coolant loop penetrations are the pressurizer surge line in one reactor vessel outlet pipe; the four safety injection inlet nozzles, one in each reactor vessel inlet pipe; two outlet nozzles to the Shutdown Cooling System, one in each reactor vessel outlet pipe; two pressurizer spray nozzles; vent and drain connections; and sample and instrument connections.

Overpressure protection for the reactor coolant pressure boundary is provided by three spring-loaded ASME Code pressurizer safety valves connected to the

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Reference E.3

NUREG-1432
Vol 1, Rev. 3.0

Standard Technical Specifications Combustion Engineering Plants

Specifications

Manuscript Completed: March 2004
Date Published: June 2004

Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 10 gpm identified LEAKAGE,

d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs), and

e. [720] gallons per day primary to secondary LEAKAGE through any one SG.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	-----NOTE----- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	 72 hours
SR 3.4.13.2	Verify SG tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS LEAKAGE detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analysis radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 50 or the staff approved licensing basis (i.e., a small fraction of these limits).

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

BASES

LCO (continued)

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident analysis. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

The [720] gallon per day limit on primary to secondary LEAKAGE through any one SG allocates the total 1 gpm allowed primary to secondary LEAKAGE equally between the two generators.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

BASES

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists or if unidentified, identified, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, [and RCP seal injection and return flows]). Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [15].

other entity; and (2) any legal successor, representative, agent, or agency of the foregoing.

Price-cap regulation means the system of rate regulation in which a rate regulatory authority establishes rates electric generator may charge its customers that are based on a specified maximum price of electricity.

Procurement document means, for the purposes of § 50.55(e) of this chapter, a contract that defines the requ which facilities or basic components must meet in order to be considered acceptable by the purchaser.

Produce, when used in relation to special nuclear material, means (1) to manufacture, make, produce, or refin nuclear material; (2) to separate special nuclear material from other substances in which such material may be or (3) to make or to produce new special nuclear material.

Production facility means:

(1) Any nuclear reactor designed or used primarily for the formation of plutonium or uranium-233; or

(2) Any facility designed or used for the separation of the isotopes of plutonium, except laboratory scale faciliti or used for experimental or analytical purposes only; or

(3) Any facility designed or used for the processing of irradiated materials containing special nuclear material, laboratory scale facilities designed or used for experimental or analytical purposes, (ii) facilities in which the on nuclear materials contained in the irradiated material to be processed are uranium enriched in the isotope U-23 plutonium produced by the irradiation, if the material processed contains not more than 106 grams of plutoniu of U-235 and has fission product activity not in excess of 0.25 millicuries of fission products per gram of U-235 facilities in which processing is conducted pursuant to a license issued under parts 30 and 70 of this chapter, o regulations of an Agreement State, for the receipt, possession, use, and transfer of irradiated special nuclear m which authorizes the processing of the irradiated material on a batch basis for the separation of selected fission and limits the process batch to not more than 100 grams of uranium enriched in the isotope 235 and not more grams of any other special nuclear material.

Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:

(1) Part of the reactor coolant system, or

(2) Connected to the reactor coolant system, up to and including any and all of the following:

(i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,

(ii) The second of two valves normally closed during normal reactor operation in system piping which does not primary reactor containment,

(iii) The reactor coolant system safety and relief valves.

For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and inc outermost containment isolation valve in the main steam and feedwater piping.

Research and development means (1) theoretical analysis, exploration, or experimentation; or (2) the extensio investigative findings and theories of a scientific or technical nature into practical application for experimental a demonstration purposes, including the experimental production and testing of models, devices, equipment, ma processes.

Responsible officer means, for the purposes of § 50.55(e) of this chapter, the president, vice-president, or othe in the organization of a corporation, partnership, or other entity who is vested with executive authority over ac subject to this part.

NEI 99-01
Rev. 4
(NUMARC/NESP-007)

Methodology for Development of Emergency Action Levels

January 2003

- The (site-specific) value for the "Potential Loss" EAL corresponds to the top of the active fuel. For sites using CSFSTs, the "Potential Loss" EAL is defined by the Core Cooling - ORANGE path. The (site-specific) value in this EAL should be consistent with the CSFST value.

5. Containment Radiation Monitoring

The (site-specific) reading is a value which indicates the release of reactor coolant, with elevated activity indicative of fuel damage, into the containment. The reading should be calculated assuming the instantaneous release and dispersal of the reactor coolant noble gas and iodine inventory associated with a concentration of 300 µCi/gm dose equivalent I-131 into the containment atmosphere. Reactor coolant concentrations of this magnitude are several times larger than the maximum concentrations (including iodine spiking) allowed within technical specifications and are therefore indicative of fuel damage. This value is higher than that specified for RCS barrier Loss EAL #4. Thus, this EAL indicates a loss of both the fuel clad barrier and a loss of RCS barrier.

There is no "Potential Loss" EAL associated with this item.

6. Other (Site-Specific) Indications

This EAL is to cover other (site-specific) indications that may indicate loss or potential loss of the Fuel Clad barrier, including indications from containment air monitors or any other (site-specific) instrumentation.

7. Emergency Director Judgment

This EAL addresses any other factors that are to be used by the Emergency Director in determining whether the Fuel Clad barrier is lost or potentially lost. In addition, the inability to monitor the barrier should also be incorporated in this EAL as a factor in Emergency Director judgment that the barrier may be considered lost or potentially lost. (See also IC SG1, "Prolonged Loss or All Offsite Power and Prolonged Loss of All Onsite AC Power", for additional information.)

RCS BARRIER EXAMPLE EALs: (1 or 2 or 3 or 4 or 5 or 6)

The RCS Barrier includes the RCS primary side and its connections up to and including the pressurizer safety and relief valves, and other connections up to and including the primary isolation valves.

1. Critical Safety Function Status

This EAL is for PWRs using Critical Safety Function Status Tree (CSFST) monitoring and functional restoration procedures. For more information, please refer to Section 3.9 of this report. RED path indicates an extreme challenge to the safety function derived from appropriate instrument readings, and these CSFs indicate a potential loss of RCS barrier.

There is no "Loss" EAL associated with this item.

2. RCS Leak Rate

The "Loss" EAL addresses conditions where leakage from the RCS is greater than available inventory control capacity such that a loss of subcooling has occurred. The loss of subcooling is the fundamental indication that the inventory control systems are inadequate in maintaining RCS pressure and inventory against the mass loss through the leak.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through steam generators,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and Reactor Coolant System Pressure Isolation Valve leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. (Note, however, that this may lead to ACTION requirements for systems involved.) Otherwise, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity at least once per 12 hours.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the containment sump inventory and discharge at least once per 12 hours,

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode,

d. Monitoring the reactor head flange leakoff system at least once per 24 hours, and

e. Verifying each Reactor Coolant System Pressure Isolation Valve leakage (Table 3.4.6-1) to be within limits:

   1. Prior to entering MODE 2 after refueling,

   2. Prior to entering MODE 2, whenever the plant has been in COLD SHUTDOWN for 7 days or more if leakage testing has not been performed in the previous 9 months,

   3. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

   4. The provision of Specification 4.0.4 is not applicable for entry into MODE 3 or 4.

f. Whenever integrity of a pressure isolation valve listed in Table 3.4.6-1 cannot be demonstrated the integrity of the remaining check valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in each high pressure line having a leaking valve shall be recorded daily.

TABLE 3.4 6-1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

Check Valve No.
V3227
V3123
V3217
V3113
V3237
V3133
V3247
V3143
V3124
V3114
V3134
V3144

NOTES

(a) Maximum Allowable Leakage (each valve):

1. Leakage rates less than or equal to 1.0 gpm are acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount the reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(c) Minimum test differential pressure shall not be less than 150 psid.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through steam generators and 720 gallons per day through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage (except as noted in Table 3.4-1) at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With RCS leakage alarmed and confirmed in a flow path with no flow indication, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere gaseous and particulate radioactivity monitor at least once per 12 hours.

b. Monitoring the containment sump inventory and discharge at least once per 12 hours.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

d. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve check valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve,

d. Following valve actuation due to automatic or manual action or flow through the valve:

    1. Within 24 hours by verifying valve closure, and

    2. Within 31 days by verifying leakage rate.

4.4.6.2.3 Each Reactor Coolant System Pressure Isolation Valve motor-operated valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit;

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

Check Valve No.		Motor Operated Valve No.
V3217	V3525	V3480
V3227	V3524	V3481
V3237	V3527	V3652
V3247	V3526	V3651
V3259		
V3258		
V3260		
V3261		
V3215		
V3225		
V3235		
V3245		

NOTES

(a) Maximum Allowable Leakage (each valve):

1. Except as noted below leakage rates greater than 1.0 gpm are unacceptable.

2. For motor-operated valves (MOVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount the reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. For motor-operated valves (MOVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(c) Minimum test differential pressure shall not be less than 200 psid.

NUREG-0818

Emergency Action Levels for Light Water Reactors

Draft Report for Comment

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Division of Emergency Preparedness
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

EXAMPLE INITIATING CONDITIONS: NOTIFICATION OF UNUSUAL EVENT

1. Emergency Core Cooling Systems (ECCS) initiated and discharge to vessel.
2. Radiological effluent technical specification limits exceeded.
3. Fuel damage indication. Examples:
   a. High offgas at BWR air ejector monitor (greater than 500,000 uci/sec; corresponding to 16 isotopes decayed to 30 minutes; or an increase of 100,000 uci/sec within a 30 minut time period)
   b. High coolant activity sample (e.g., exceeding coolant technical specifications for iodine spike)
   c. Failed fuel monitor (PWR) indicated increase greater than 0.1% equivalent fuel failures within 30 minutes.
4. Abnormal coolant temperature and/or pressure or abnormal fuel temperatures outside of technical specification limits.
5. Exceeding either primary/secondary leak rate technical specification or primary system leak rate technical specification.
6. Failure of a safety or relief valve in a safety related system to close following reduction of applicable pressure.
7. Loss of offsite power or loss of onsite AC power capability.
8. Loss of containment integrity requiring shutdown by technical specifications.
9. Loss of engineered safety feature or fire protection system function requiring shutdown be technical specifications (e.g., because of malfunction, personnel error or procedural inadequacy).
10. Fire within the plant lasting more than 10 minutes.
11. Indications or alarms on process or effluent parameters not functional in control room to an extent requiring plant shutdown or other significant loss of assessment or communication capability (e.g., plant computer, Safety Parameter Display System, all meteorological instrumentation).
12. Security threat or attempted entry or attempted sabotage.
13. Natural phenomenon being experienced or projected beyond usual levels:
    a. Any earthquake felt in-plant or detected on station seismic instrumentation

b. 50 year flood or low water, tsunami, hurricane surge, seiche
c. Any tornado on site
d. Any hurricane.

14. Other hazards being experienced or projected:
    a. Aircraft crash on-site or unusual aircraft activity over facility
    b. Train derailment on-site
    c. Near or onsite explosion
    d. Near or onsite toxic or flammable gas release.
    e. Turbine rotating component failure causing rapid plant shutdown.
15. Other plant conditions exist that warrant increased awareness on the part of a plant operating staff or State and/or local offsite authorities or require plant shutdown under technical specification requirements or involve other than normal controlled shutdown (e.g., cooldown rate exceeding technical specification limits, pipe cracking found during operation).
16. Transportation of contaminated injured individual from site to offsite hospital.
17. Rapid depressurization of PWR secondary side.

acceptable value of subcooling margin will differ depending on whether the reactor is at power, hot standby, or tripped.

Initiating Condition No. 5

Exceeding either primary/secondary leak rate technical specification or primary system leak rate technical specification.

Draft EALs

Primary to secondary leak rate greater than 1 gpm total for more than four hours or greater than 500 gpm per steam generator as identified by daily RCS leakage evaluation; or

Primary system leak rate greater than those specified in Technical Specification 3.4.6.2 as identified by daily RCS leakage evaluation.

1. > 0 pressure boundary leakage
2. > 1 gpm unidentified for more than 4 hours
3. > 10 gpm identified RCS leakage for more than 4 hours
4. > 30 gpm controlled leakage (2235 ± 20 psig) for more than 4 hours.

Discussion

The response is adequate.

Initiating Condition No. 6

Failure of a safety or relief valve in a safety related system to close following reduction of applicable pressure.

Draft EALs

Pressurizer or steam generator safety or relief valve opens and then fails to reset as indicated by:

1. Pressurizer relief valve position light indicates open; or
   Pressurizer safety valve position indicator reads greater than 1%; or
   Valid accoustical monitor indication.

NUREG-0654
FEMA-REP-1
Rev. 1

Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

Manuscript Completed: October 1980
Date Published: November 1980

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Federal Emergency Management Agency
Washington D.C. 20472

Class	Licensee Actions	State and/or Local Offsite Authority Actions
NOTIFICATION OF UNUSUAL EVENT Class Description Unusual events are in process or have occurred which indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs. Purpose Purpose of offsite notification is to (1) assure that the first step in any response later found to be necessary has been carried out, (2) bring the operating staff to a state of readiness, and (3) provide systematic handling of unusual events information and decisionmaking.	1. Promptly inform State and/or local offsite authorities of nature of unusual condition as soon as discovered. 2. Augment on-shift resources as needed 3. Assess and respond 4. Escalate to a more severe class, if appropriate or 5. Close out with verbal summary to offsite authorities; followed by written summary within 24 hours	1. Provide fire or security assistance if requested 2. Escalate to a more severe class, if appropriate 3. Stand by until verbal closeout

U.S. ATOMIC ENERGY COMMISSION

May 1973

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.45

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS

A. INTRODUCTION

General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary," of Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants," requires that means be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. This guide describes acceptable methods of implementing this requirement with regard to the selection of leakage detection systems for the reactor coolant pressure boundary. This guide applies to light-water-cooled reactors. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

The safety significance of leaks from the reactor coolant pressure boundary (RCPB) can vary widely depending on the source of the leak as well as the leakage rate and duration. Therefore, the detection and monitoring of leakage of reactor coolant into the containment area is necessary. In most cases, methods for separating the leakage from an identified source from the leakage from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action should a leak be detrimental to the safety of the facility. Identified leakage is: (1) leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank, or (2) leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of unidentified leakage monitoring systems or not to be from a flaw in the RCPB. Unidentified leakage is all other leakage.

Leakage Separation

A limited amount of leakage is expected from the RCPB and from auxiliary systems within the containment such as from valve stem packing glands, circulating pump shaft seals, and other equipment that cannot practically be made 100% leaktight. The reactor vessel closure seals and safety and relief valves should not leak significantly; however, if leakage occurs via these paths or via pump and valve seals, it should be detectable and collectable and, to the extent practical, isolated from the containment atmosphere so as not to mask any potentially serious leak should it occur. These leakages are known as "identified leakage" and should be piped to tanks or sumps so that the flow rate can be established and monitored during plant operation.

Uncollected leakage to the containment atmosphere from sources such as valve stem packing glands and other sources that are not collected increases the humidity of the containment. The moisture removed from the atmosphere by air coolers together with any associated liquid leakage to the containment is known as "unidentified leakage" and should be collected in tanks or sumps where the flow rate can be established and monitored during plant operation. A small amount of unidentified leakage may be impractical to eliminate, but it should be reduced to a small flow rate, preferably less than one gallon per minute (gpm), to permit the leakage detection systems to detect positively and rapidly a small increase in flow rate. Thus a small unidentified leakage rate that is of concern will not be masked by a larger acceptable identified leakage rate.

Substantial intersystem leakage from the RCPB to other systems across passive barriers or valves is not expected. However, should such leakage occur, it may not be detectable through the above-mentioned detection systems, and other alarm and detection methods should be employed. For example, steam generator leakage in pressurized water reactors (PWR's) should be monitored to detect tube or tube sheet leaks.

Acceptable Detection Methods

Although monitoring of both identified and unidentified leakage is important, effective systems for detecting and locating unidentified leakage are also needed. The following paragraphs describe some acceptable detection methods.

In addition to monitoring flow rate changes to tanks and sumps for liquid collection, other methods should be included to indicate when and where coolant is released to the containment atmosphere. For example, these additional detection methods would indicate and/or monitor changes in:

a. airborne particulate radioactivity,
b. airborne gaseous radioactivity,
c. containment atmosphere humidity,
d. containment atmosphere pressure and temperature,
e. condensate flow rate from air coolers.

Since intersystem leakage does not release reactor coolant to the containment atmosphere, detection methods should include monitoring of water radioactivity in the connected systems where the systems flows through the containment boundary, and monitoring of airborne radioactivity where such systems are vented outside the containment boundary. Another important method of obtaining indications of uncontrolled or undesirable intersystem flow would be the use of a water inventory balance, designed to provide appropriate information such as abnormal water levels in tanks and abnormal water flow rates.

Potential discharges from closed safety and relief valves are usually piped to tanks or water pools and considered part of identified leakage. Temperature sensors in the discharge path of safety and relief valves or flow meters in the leak-off lines would provide an acceptable method of signaling small leakage from these valves.

While the above-mentioned leakage detection systems reflect the present state of technology, it is recognized that other detection methods may be developed and used in order to obtain operating experience with them. Among such methods are sonic indicators and moisture sensitive tapes applied to RCPB component parts. Because of the potential importance of early leak detection in the prevention of accidents, continued improvements in leakage detection and locating techniques should be sought.

It is not necessary that all of the above-mentioned leakage detection methods or systems be employed in a specific nuclear power plant. However, since the methods differ in sensitivity and response time, prudent selection of detection methods should include sufficient systems to assure effective monitoring during periods when some detection systems may be ineffective or inoperable. Some of these systems should serve as early alarm systems signaling the operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required.

Detector Sensitivity

It is essential that leakage detection systems have the capability to detect significant RCPB degradation as soon after occurrence as practical to minimize the potential for a gross boundary failure. It is possible that some cracks might develop and penetrate the RCPB wall, exhibit very slow growth, and afford ample time for a safe and orderly plant shutdown after a leak is detected. On the other hand, leakage such as that resulting from stress-assisted corrosion in stainless steel or from a flaw at a high fatigue point in the RCPB would demand rapid detection and probable plant shutdown. Therefore, an early warning signal is necessary to permit proper evaluation of all unidentified leakage.

Industry practice has shown that water flow rate changes of from 0.5 to 1.0 gpm can readily be detected in containment sumps by monitoring changes in sump water level, in flow rate, or in the operating frequency of pumps. Sumps and tanks used to collect unidentified leakage and air cooler condensate should be instrumented to alarm for increases of from 0.5 to 1.0 gpm in the normal flow rates. This sensitivity would provide an acceptable performance for detecting increases in unidentified liquid leakage by this method.

An increase in humidity of the containment atmosphere would indicate release of water vapor to the containment. Dew point temperature measurements can be used to monitor humidity levels of the containment atmosphere. A 1° increase in dew point is well within the sensitivity range capability of available instruments. Since the humidity level is influenced by several factors, a quantitative evaluation of an indicated leakage rate may be questionable and should be compared to observed increases in liquid flow from sumps and condensate flow from air coolers. Humidity level monitoring is considered most useful as an alarm or indirect indicating device to alert the operator to a potential problem.

Reactor coolant normally contains sources of radiation which, when released to the containment, can be detected by the monitoring systems. However, reactor coolant radioactivity should be low during initial reactor startup and for a few weeks thereafter until activated corrosion products have been formed and fission products become available from failed fuel elements; during this period, radioactivity monitoring instruments may be of limited value in providing an early warning of very small leaks in the RCPB. Instrument sensitivities of 10^{-9} μCi/cc radioactivity for air particulate monitoring and of 10^{-6} μCi/cc radioactivity for radiogas monitoring are practical for these leakage detection systems. Radioactivity monitoring systems should be included for every plant (especially particulate activity monitoring) because of their sensitivity and rapid response to leaks from the RCPB.

Air temperature and pressure monitoring methods may also be used to infer RCPB leakage to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCPB leakage into the containment. The accuracy and relevance of temperature and pressure measurements is a function of containment free volume and detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable energy releases to the containment.

While the concern about instrument sensitivity applies to the lower range of service for which the instruments are selected, the upper instrument range limits should be established to prevent exceeding the saturation limits of instruments, thus making them useless as indicators of containment conditions.

Detector Response Time

The need to evaluate the severity of an alarm or indication is important to the operators, and the ability to compare with indications from other systems is necessary. The system response time should therefore be included in the functional requirements for leakage detection systems. Except for the limitations during the initial few weeks of plant operation as discussed previously, all detector systems should respond to a one gpm, or its equivalent, leakage increase in one hour or less. Multiple instrument locations in monitored areas should be utilized if necessary to assure that the transport delay time of the leakage effluent from its source to the detector or instrument location will yield an acceptable overall response time. A useful technique in identifying the general location of a leakage area is the placing of several sensors within the containment area and observing differences in response from the sensors, and this technique should be used to satisfy this requirement of General Design Criterion 30.

In analyzing the sensitivity of leak detection systems using airborne particulate or gaseous radioactivity, a realistic primary coolant radioactivity concentration assumption should be used. The expected values used in the plant environmental report would be acceptable.

Signal Correlation and Calibration

It is important to be able to associate a signal or indication of a change in the normal operating conditions with a quantitative leakage flow rate. Except for flow rate or level change measurements from tanks, sumps, or pumps, signals from other leakage detection systems do not provide information readily convertible to a common denominator. Approximate relationships converting these signals to units of water flow should be formulated to assist the operator in interpreting signals. Since operating conditions may influence some of the conversion procedures, the procedures should be revised during such periods. To assure the continued reliability of the leakage detection systems, the equipment should comply with Paragraph 4.10 of IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations,"[1] for tests and calibration.

Seismic Qualification

Since nuclear power plants may be operating at the time an earthquake occurs and may continue to operate after earthquakes, it is prudent to require the leakage detection systems to function under the same conditions. If a seismic event comparable to a safe shutdown earthquake (SSE) occurs, it would be important for the operator to assess the condition within the containment quickly. The proper functioning of at least one leakage detection system would assist in evaluating the seriousness of the condition within the containment in the event leakage has developed in the RCPB. The airborne particulate radioactivity monitoring equipment has the desirable sensitivity to indicate RCPB leakage, and it should be included for all plants. Components for the airborne particulate radioactivity equipment should be qualified to function through the SSE.

C. REGULATORY POSITION

The source of reactor coolant leakage should be identifiable to the extent practical. Reactor coolant pressure boundary leakage detection and collection systems should be selected and designed to include the following:

1. Leakage to the primary reactor containment from identified sources should be collected or otherwise isolated so that:
   a. the flow rates are monitored separately from unidentified leakage, and
   b. the total flow rate can be established and monitored.

2. Leakage to the primary reactor containment from unidentified sources should be collected and the flow

[1] Copies may be obtained from the Institute of Electrical and Electronics Engineers, United Engineering Center, 345 East 47th Street, New York, N.Y. 10017.

rate monitored with an accuracy of one gallon per minute (gpm) or better.

3. At least three separate detection methods should be employed and two of these methods should be (1) sump level and flow monitoring and (2) airborne particulate radioactivity monitoring. The third method may be selected from the following:

a. monitoring of condensate flow rate from air coolers,

b. monitoring of airborne gaseous radioactivity.

Humidity, temperature, or pressure monitoring of the containment atmosphere should be considered as alarms or indirect indication of leakage to the containment.

4. Provisions should be made to monitor systems connected to the RCPB for signs of intersystem leakage. Methods should include radioactivity monitoring and indicators to show abnormal water levels or flow in the affected area.

5. The sensitivity and response time of each leakage detection system in regulatory position 3. above employed for unidentified leakage should be adequate to detect a leakage rate, or its equivalent, of one gpm in less than one hour.

6. The leakage detection systems should be capable of performing their functions following seismic events that do not require plant shutdown. The airborne particulate radioactivity monitoring system should remain functional when subjected to the SSE.

7. Indicators and alarms for each leakage detection system should be provided in the main control room. Procedures for converting various indications to a common leakage equivalent should be available to the operators. The calibration of the indicators should account for needed independent variables.

8. The leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operation.

9. The technical specifications should include the limiting conditions for identified and unidentified leakage and address the availability of various types of instruments to assure adequate coverage at all times.