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

DEC 2 6 1979

Mr. G. E. Liebler, Chairman
Combustion Engineering Owners' Group
Florida Power & Light Company
P. O. Box 013100
Miami, Florida 33101

Dear Mr. Liebler:

SUBJECT: EVALUATION OF OPERATOR GUIDELINES FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS IN C-E DESIGNED OPERATING PLANTS

Our letter of June 5, 1979 (Robert W. Reid to all operating Combustion Engineering plants) requested that operating plants with C-E-designed reactors develop guidelines for the preparation of operating procedures to cope with small-break LOCA's. In response to this request, the C-E Owners' Group submitted report CEN-114-P (Amendment 1P) which included these guidelines. In response to our requests for additional information and to issues raised during our meeting of October 30, 1979, the guidelines were subsequently modified. The modified guidelines were submitted by your letter dated November 8, 1979. In my letter to you dated November 14, 1979, we approved the modified guidelines for all C-E operating plants except for a plant having high pressure safety injection pumps with a 2400 psi shutoff head.

In your letter of December 13, 1979 (see Enclosure 1) you provided modified guidelines for this class of plant. Subsequent to that, we held discussions with members of the C-E Owners' Group and C-E to clarify certain matters. We have now completed our review of the modified guidelines. Our supplemental evaluation is provided as Enclosure 2 to this letter. The supplemental evaluation in Enclosure 2, together with the evaluation provided in my letter to you dated November 14, 1979, comprise the bases for our approval of the guidelines for this class of plant. The November 14, 1979 letter is provided as Enclosure 3 to this letter.

The November 14, 1979 letter contains a number of provisions which licensees are required to meet in implementing the guidelines. These provisions are equally applicable to those licensees that develop procedures from the revised guidelines.

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All licensees with C-E-designed reactors are expected to proceed with the development of small break LOCA emergency procedures and operator training. As indicated on Page 5 of Enclosure 6 to the Darrell G. Eisenhut letter dated September 13, 1979 to all operating nuclear power plants, these procedures and related operator training are to be implemented by December 31, 1979.

Sincerely,

D. F. Ross, Jr.

D. F. Ross, Jr., Director
Bulletins and Orders Task Force

Enclosures:
As stated

cc: See attached lists

ENCLOSURE 1

December 13, 1979

POOR ORIGINAL

Dr. Denwood F. Ross, Jr.
Director
Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission

Subject: Additional Post-LOCA Guidance for Plants with High Pressure Safety Injection Pumps with a 2400 psi Shut-off Head

Reference: NRC letter from Dr. D. F. Ross, Jr. to Mr. G. E. Liebler, dated November 14, 1979

Dear Dr. Ross:

Your referenced letter forwarded the NRC evaluation of the LOCA guideline for CE designed plants. That evaluation concluded that the guidelines are acceptable for CE operating plants having high pressure safety injection pumps with shut-off heads less than 1600 psi. However, the NRC has not yet determined that the guidelines are acceptable for a plant (Maine Yankee) having high pressure safety injection pumps with shutoff heads of 2400 psi.

The NRC is concerned with potential events in which water could be discharged through the safety valves while the operator is attempting to achieve a RCS fluid condition of at least 50° below saturation. The question is whether or not 50° of subcooling can be achieved, at which point the operator is allowed to stop high pressure flow to the RCS, before the pressure in the RCS reaches the setpoint of the safety valves and water is discharged through these valves. The shutoff head of the Maine Yankee pump (2425 psi) is below the setpoint of the safety valves (2500 psi) but is above the setpoint of the PORV (2400 psi) and this needs to be addressed.

To evaluate the NRC concern, representative non LOCA events which depressurize the RCS were studied. The events chosen were a failure of the reactor coolant pressure regulating system and a steam line break. Non-LOCA events result in the highest RCS repressurization due to high pressure pump action. These events essentially represent a "zero break" LOCA case. In this study it was assumed that all four RCP's were tripped following SIS actuation and that the resulting flow coastdown causes loss of pressurizer sprays. The study confirmed that 50° of subcooling is achieved prior to reaching the setpoint of the PORV. If no operator action is taken the high pressure pumps could cause the PORV's to lift.

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There is approximately 5 minutes between the point in time that 50° sub-cooling is reached and the setpoint of the PORV is reached. Additionally, when the PORV setpoint (2400 psi) is reached, the pressurizer is not water solid. The study also indicates that there is an additional 5 minutes before the pressurizer is filled solid assuming two HPSI pumps are operating.

This is judged to be sufficient time frame for the operator to take action. The action the operator takes in this situation is as follows:

Preferred Action - Terminate high pressure pump flow to the reactor coolant system.

Alternative Action - Prevent operation of the PORV's by:

(a) shutting down stream block valves.

or

(b) position PORV's control switch to prevent automatic opening.

As a result of the above evaluation it has been concluded that an additional precaution should be added to the CE Post LOCA Guidelines in order to incorporate applicability to the Maine Yankee Plant. That precaution forms the attachment to this letter. The addition of this precaution should result in a determination that the LOCA guidelines are acceptable for developing operating procedures for Maine Yankee.

If you should have any questions regarding this guidance, please feel free to contact me or Mr. R. T. Harris of our Technical Advisary Committee at (203) 666-6911, extension 5519.

Very truly yours,

C-E OWNERS GROUP

Robert T. Harris for
George E. Liebler
Chairman

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ADDITIONAL GUIDANCE FOR PLANTS WITH HIGH PRESSURE SAFETY INJECTION PUMPS WITH A 2400 PSI SHUT-OFF HEAD

Add the additional precaution to the Post LOCA Guidelines as follows:

12. An SIAS can be generated by events other than a LOCA, such as a failure of the reactor coolant pressure regulating system. Continued operation of high head-high pressure injection pumps can cause the RCS to repressurize to the setpoint of the PORV's. Opening of the PORV's should be prevented by:

    (a) operating the SIS to maintain RCS pressure below the PORV setpoint (2400 psi) while maintaining at least 50°F subcooling.

    or

    (b) position the PORV control switch to prevent automatic opening of the PORV's.

    or

    (c) shut the PORV block valves to negate consequences of PORV's opening.

EVALUATION OF SMALL-BREAK LOCA GUIDELINES FOR C-E OPERATING PLANTS HAVING SAFETY INJECTION PUMPS WITH SHUTOFF HEADS GREATER THAN 1600 PSI

INTRODUCTION

By letter dated November 14, 1979, we approved the guidelines for developing small-break LOCA procedures in C-E operating plants. This approval was limited to plants having safety injection (SI) pumps with shutoff heads less than 1600 psi; therefore, to support the implementation of these guidelines at plants having SI pumps with shutoff heads greater than 1600 psi, (i.e., Maine Yankee), the C-E Owners Group submitted additional information in a letter dated December 13, 1979. In addition, on December 13 and 14, 1979, we held discussions with C-E personnel regarding our concerns associated with plants having SI pumps with high shutoff pressures.

EVALUATION

To evaluate the potential of lifting the PORVs in C-E designed plants, all of which have a setpoint of 2400 psi, prior to satisfying the 50°F subcooling criterion at plants having SI pumps with a 2425 psi shutoff head (i.e., Maine Yankee), two non-LOCA events were analyzed: (i) failure of the pressure regulating system, and (ii) a steam line break. The maximum calculated hot leg temperature for these events was 540°F. Since this temperature is significantly below 612°F, the saturation temperature at 2400 psi with a 50° subcooling margin, our subcooling criterion should be satisfied prior to lifting the PORVs. This calculated temperature of 540°F was based on the dynamic conditions prevailing during the pressurizer refill portion of the transients.

On December 13 and 14, 1979, the staff discussed a more conservative steady state analysis with C-E personnel wherein no credit is taken for the dynamic effects of cold feedwater and SI flow. Such an analysis simply considers natural circulation in the primary system transferring heat to the steam generator (SG) whose temperature corresponds to the saturation pressure of the SG safety valves. C-E stated that under these conditions, their analyses show that the maximum hot leg temperature would be 580°F. Since this temperature is also below the 612°F cited above, the 50°F subcooling criterion would also be met without exceeding 2400 psi, the PORV setpoint.

CONCLUSIONS

Based on the results of the analyses that show that the 50°F criterion will be met prior to lifting the PORVs, we find that the LOCA guidelines are acceptable for C-E plants having SI pumps with shutoff heads of 2425 psi (i.e., Maine Yankee). This approval, however, is contingent upon receiving documentation from the C-E Owners Group of the analyses showing that the 50°F subcooling criterion can be met without exceeding 2400 psi.

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

NOV 1 1979

Mr. G. E. Liebler, Chairman
Combustion Engineering Owners Group
Florida Power & Light Company
P. O. Box 013100
Miami, Florida 33101

POOR ORIGINAL

Dear Mr. Liebler:

SUBJECT: EVALUATION OF OPERATOR GUIDELINES FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS IN C-E DESIGNED OPERATING PLANTS

Our letter of June 5, 1979 (Robert W. Reid to all operating Combustion Engineering plants) requested that operating plants with C-E designed reactors develop guidelines for the preparation of operating procedures to cope with small-break LOCA's. In response to this request, the C-E Owners Group submitted report CEN-114-P (Amendment 1P) which included said guidelines. In response to our requests for additional information and to issues raised during our meeting of October 30, 1979, the guidelines were subsequently modified. The modified guidelines were submitted by your letter to D. F. Ross dated November 8, 1979. We have completed our review of the modified guidelines, and are attaching hereto as Enclosure 1 a copy of our evaluation.

As stated in our evaluation, we have concluded that the guidelines submitted by your November 8, 1979 letter are acceptable for use in developing operating procedures to cope with small-break LOCA's in C-E operating plants having high-pressure safety injection pumps with shut-off heads less than 1600 psi. Although the guidelines were based on a reference plant having 200 psi safety injection tanks and 1300 psi high-pressure safety injection pumps, you have stated that they are applicable to all operating C-E plants, including those with 600 psi safety injection tanks and those with 2400 psi high-pressure safety injection pumps. However, we have not as yet determined that the guidelines are acceptable for a plant having high-pressure safety injection pumps with a 2400 psi shut-off head. Our concern is related to the potential events in which water could be discharged through the safety valves while the operator is attempting to reach a condition of at least 50° F below saturation. A copy of the approved guidelines, subject to acceptably incorporating those revisions required by Enclosure 1, is attached hereto as Enclosure 2.

Those licensees with C-E designed reactors for which these guidelines are approved may now proceed with the development of small-break LOCA emergency procedures and operator training. In developing these procedures, each licensee must account for the effects of specific design characteristics at its plant. As indicated on Page 5 of Enclosure 6 to the Darrell G. Eisenhut letter dated September 13, 1979 to all operating nuclear power plants, these procedures and related operator training are to be implemented by December 31, 1979.

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In implementing these procedures, each licensee shall provide:

(1) The instrument uncertainties involved with HPI termination criteria to indicate that the criteria will assure subcooled conditions.

(2) Adequate assurance that the HPSI pumps will not be run deadheaded in the recirculation mode and that minimum flow requirements will be met.

(3) An indication of the typicality of the analyses documented in CEN-114-P (Amendment 1P) and in the modified guidelines shown in Enclosure 2 relative to its own plant.

Licensees will also be required to implement emergency procedures covering the extended loss of all feedwater, (including pressure vessel integrity considerations), and to revise emergency procedures for initiating and monitoring natural circulation, including provisions for plant cooldown. These procedures will be based on guidelines which the C-E Owners Group are developing under "inadequate core cooling."

As part of our audit program, we expect to examine the procedures at a lead C-E operating plant initially, and at other C-E operating plants at a later date to assure that the procedures were developed in accordance with the approved guidelines. We also plan to check out some of the procedures at a C-E simulator on a schedule to be developed later. It should be noted however, that our audit program need not impede progress toward implementing the procedures and associated training by December 31, 1979.

Sincerely,

D. F. Ross, Jr., Director
Bulletins & Orders Task Force

Enclosures:
As stated
cc: See attached lists

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ENCLOSURE 1

90022201

Evaluation of Combustion Engineering Post-LOCA Operating Guidelines

Introduction

By letter dated June 5, 1979, the staff requested that all operating CE plants provide guidelines for the preparation of operational procedures for the recovery of plants following small LOCA's. The guidelines were to cover both short-term and long-term situations and follow through to a stable condition. Recognition of the event, precautions, actions, and prohibited actions were to be included also. CE submitted CEN-114-P-(NP), "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems" in July, 1979 and CEN-115-P(NP), "Response to NRC IE Bulletin 79-06C Items 2 and 3 for Combustion Engineering Nuclear Steam Supply Systems" in August, 1979. CEN-114-P(NP) was submitted in response to our request for information while CEN-115-P(NP) revised this response to account for the impact of RCP operating requirements.

Summary Description: CE Post-LOCA Operating Guidelines

The guideline submitted by CE is preceded by a bases section which supplies background material for the information presented in the guideline. The guideline itself is split into four sections: Symptoms, Immediate Actions, Follow-Up Actions, and Precautions.

The Symptoms are a list of indications which an operator is expected to utilize in confirming that a small break loss-of-coolant accident has occurred. Low pressurizer pressure, high containment sump level, high containment pressure or temperature, safety injection actuation, and high or low pressurizer level are among the symptoms provided to the operator to assist in the identification of this accident. A diagnostics chart has been appended to the LOCA guidelines to clarify symptoms and to channel the operator's actions into the correct procedure.

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<u>Immediate Actions</u> are those actions which are required to place the plant in a safe condition. These steps are distinguished from subsequent procedural steps by a requirement for memorization. An operator must know these steps without reference to a procedure, thereby ensuring that there is no delay in achieving a safe condition. The guidelines require that the reactor be tripped; standard post-trip actions be carried out (plant specific); safety injection be initiated (if not automatically actuated); reactor coolant pumps be tripped after SIAS actuation on low RCS pressure; auxiliary feedwater flow be established if main feedwater is not available; verification that the CIAS and SIAS signals have properly actuated; the SIS be operated to maintain a 50°F subcooling margin and indicated pressurizer level; and the break be located and isolated if possible.

<u>Follow-Up Actions</u> are actions required to place the plant in a stable condition. The previous procedural steps (Immediate Actions) ensured that the reactor was in a safe condition, that the core remains covered by ECCS operation, and that escaping radioactivity is isolated by CIAS. The next steps are aimed at bringing the plant to a lower mode of operation, cold shutdown. The Follow-Up Actions require a plant cooldown within one-hour using the steam dumps or turbine bypass system. The cooldown is continued via a number of alternative paths such as long-term recirculation, initiation of shutdown cooling, continued use of the steam dumps and emergency feed, or, as a last resort, opening of the power operated relief valves.

The <u>Precautions</u> section lists warnings which the operator must observe to ensure plant safety. For example, the operator is warned that pressurizer level may not always be a true indicator of fluid inventory and that primary system temperature must be monitored when establishing auxiliary feedwater to prevent excessive cooldown rates. A total of eleven Precautions have been included for implementation by the licensees in the appropriate procedural locations.

Evaluation

The NRC staff reviewed the post-LOCA operating guidelines with respect to the following critical operator actions:

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1. Reactor coolant pump trip
2. Safety injection termination criteria
3. Verification of safety systems actuation
4. Verification of a heat sink.

During our review, the staff identified modifications to be made to the guidelines to enhance the directions to the operator. These modifications were subsequently incorporated in the guidelines via revisions issued on November 8, 1979.

The criteria for tripping the reactor coolant pumps are consistent with the requirements of IE Bulletin 79-06C. All operating reactor coolant pumps are stopped after an SIAS caused by low reactor coolant system pressure and after it has been verified that the reactor has been shutdown for at least five seconds. We conclude that this criterion is acceptable subject revising "Immediate Action" item 3 of the guidelines to be consistent with the above wording.

The criterion for terminating safety injection flow is based on the establishment and maintenance of a 50°F subcooling margin along with an indication of pressurizer level. The staff concurs that these criteria are sufficient for ensuring that safety injection can be terminated without concern for detrimental voiding in the primary system. We conclude that this criterion is acceptable for those plants with low-head HPSI pumps ($<$ 1600 psi).

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As part of his immediate actions, the operator is directed to verify the reactor trip, safety injection actuation, adequate auxiliary feedwater flow (if main feedwater is not available), and containment isolation actuation. We concur that these actions are sufficient to ensure minimum safeguards and heat sink availability needed to mitigate small break LOCAs.

The staff noted that the guidelines are based on obtaining at least minimum safeguards operation to mitigate small break LOCAs. We require each licensee to extend the emergency procedures to cover the loss of all feedwater. Procedures for this degraded condition should also take into account pressure vessel integrity considerations. The Owners Group has committed to prepare guidelines for operational procedures regarding the loss of all feedwater as part of its effort on the issue of inadequate core cooling.

The staff also requires that the emergency procedures include instructions for monitoring and initiating (if lost) natural circulation for small break LOCAs where heat removal by the steam generators is required. A separate guideline has been received on natural circulation operation. The staff, upon completion of its evaluation, will require that the natural circulation guideline be appended to or referenced by the appropriate emergency procedures.

The staff requires that each licensee provide procedures for cooling down the plant under natural circulation conditions. These procedures should address boration control and monitoring, cooldown of the pressurizer, and adequate criteria for monitoring coolant system temperatures to ensure that voids do not form in the primary system which could inhibit adequate heat removal. As in

the case of loss of all feedwater, the Combustion Engineering Owners Group has committed to prepare guidelines for operational procedures regarding cooldown under natural circulation conditions as part of its effort on inadequate core cooling.

Conclusions

Based on our review, we conclude that the small-break loss-of-coolant accident operating guidelines submitted by the Combustion Engineering Owners Group on November 8, 1979 are acceptable for C-E plants having high-pressure safety injection pumps with shut-off heads 1600 psi or less. Accordingly, said guidelines can be used for developing operating procedures for coping with small-break loss-of-coolant accidents for such plants, provided that the licensees implement the requirements noted above when developing their procedures. Our acceptance of these generic guidelines notwithstanding, each licensee must account for the effects of specific plant design parameters (e.g., differences in the shut-off pressures of high-pressure safety injection pumps, differences in the design pressure of the safety injection tanks), when translating these guidelines into plant specific operating procedures.

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P.O. Box 529100
Miami, FL 33152

November 8, 1979

Dr. Denwood F. Ross, Jr.
Director
Bulletins and Orders Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Transmittal of Revised Post-LOCA Guidelines

Reference: (A) NRC letter from Dr. D. F. Ross, Jr. to Mr. G. E. Liebler, dated October 19, 1979

(B) IE Bulletin 79-06C, dated July 26, 1979

(C) NUREG-0578, July 1979

Dear Dr. Ross:

Reference A requested additional information regarding the guidelines presented in CEN-114 Revision A and CEN-115 for loss of coolant accidents (LOCA). Questions regarding those guidelines were further discussed in a meeting with the NRC staff on October 30, 1979, and a number of revisions were agreed upon. This letter transmits those revised Post-LOCA guidelines. These guidelines are being submitted for your approval on behalf of the Combustion Engineering Owners Group so that they may be incorporated into utility procedures in accordance with Reference B and the schedule presented in Reference C.

It should be noted that these guidelines do not necessarily reflect the preferred actions of our vendor, Combustion Engineering. Combustion Engineering's preferred actions remain as stated in CEN-115. The NRC staff has specifically requested that the guidelines for RCP operation be revised to incorporate the RCP operating requirements stated in IE Bulletin 79-06C (Reference A, Item I.6.E). Combustion Engineering has been unable to identify a transient analyzed in Chapter 6 or 15 of the FSAR that will result in violation of acceptance criteria, provided the RCP's are not tripped until the rods have been fully inserted for 5 seconds. The enclosed guidelines have therefore been revised to reflect the staff's request.

If you should have any questions regarding these guidelines, please feel free to contact me at (305) 552-3811.

Very truly yours,

C-E OWNERS GROUP

George E. Liebler
Chairman

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Enclosure

POST LOCA GUIDELINES

Bases for Post-LOCA Operating Guidelines

Provided below is a general description of plant responses to large and small break LOCA's. This is intended to supply background material for the information presented in the guidelines.

A small break LOCA is characterized by:

a) A slow loss of RCS pressure during the short term (10 to 30 minutes) and equilibrium pressure above * 300 psia in the long term (30 to 480 minutes) resulting from matching safety injection flow and flow from the break.

b) A loss of RCS inventory during the short term followed by a refilling of the RCS during the long term.

c) Core cooling is initially by the steam generator(s) and flow from the break and later by the shutdown cooling system. The break does not always (depending on size) provide the necessary heat removal yet depletes RCS inventory. Breaks in RCS piping less than 2 inches in diameter fall into this category. The steam generators provide cooling for forced or natural circulation of the RCS, if inventory is depleted, in a boiloff and reflux mode. The shutdown cooling system is used after the RCS has been refilled and pressure control is provided by the HPSI pumps and the charging pumps.

A general description of small break LOCA operations follows:

Initially, the plant is hot and pressurized. A small break LOCA results in a slow loss of RCS inventory and a decrease in pressure. Low pressurizer pressure initiates a SIAS which automatically actuates the SIS. The reactor is tripped. The operator stops the reactor coolant pumps. Auxiliary feedwater is established to the steam generators. Steam dump is provided manually using atmospheric dump valves or turbine bypass valves, or automatically by the steam generator dump and bypass system or by steam generator relief valves.

*This value is typical, it may vary for specific designs.

For very small breaks, the steam generators are the main heat sink, and additional heat is removed with the coolant through the break. Continued reactor coolant pump operation during this period could aid heat removal by the steam generators. However, for small hot leg breaks, reactor coolant pump operation will result in a higher two-phase mixture level in the reactor vessel and hot leg piping. Consequently, for a break in the bottom of the hot leg, the break is covered longer by two-phase mixture, causing a larger loss of water inventory from the vessel. This eventually results in a lower coolant level in the reactor vessel. The result could be a higher clad temperature and a delay in refilling the vessel. The net effect of reactor coolant pump operation during the initial period may be to increase the severity of the accident. The NRC has therefore requested that the RCP operating requirements stated in IE Bulletin 79-06C be incorporated into the guidelines for operating plants following LOCA's (NRC letter from Dr. D.F. Ross to G. E. Liebler, dated October 19, 1979). Bulletin 79-06C directed to holders of operating licenses to: "Upon reactor trip and HPI initiation caused by low reactor coolant system pressure, immediately trip all operating RCP's." This action should not result in the violation of acceptance criteria for transients or accidents in chapter 6 or 15 of the FSAR, provided the RCP's are not tripped until rods have been fully inserted for 5 seconds. This delay is to allow for the decay of the heat flux following reactor trip before reducing forced flow.

The time necessary to refill the RCS and regain control of pressure and inventory depends on break size, break location, and the number of HPSI pumps and charging pumps actuated. With only one HPSI pump activated, and a break located on the bottom of the cold leg, it may take as long as 8 hours to refill the RCS. With all injection pumps operable, the time is about 1 hour. In the period of time it takes the RCS to refill some voiding in the RCS will occur. This, condition can be recognized by indication that RCS hot leg temperature or core thermocouple temperature is equal to the saturation temperature for the existing RCS pressure. In this mode, decay heat is removed by boiling in the core and condensation in the steam generator. In addition, heat is removed by flow from the break. The operator must ensure that the SIS is providing flow to the RCS, and the steam generators are removing heat. These actions will ensure adequate core cooling and eventually a subcooled condition will be achieved. Once RCS pressure and temperature are adequately reduced, the shutdown cooling system

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is placed in operation. In the event that the feedwater supply to the steam generator is exhausted and the shutdown cooling system is inoperable, the PORV's are opened to ensure that the flow from the injection system is sufficient to cool the core. The SIS will be realigned for cold leg injection only. Core flushing is from the cold legs through the core and out the PORV.

Simultaneous hot and cold leg injection is used for both small break and large break LOCA's so the operator does not have to distinguish between them at the time when simultaneous injection is required for large breaks. (For small breaks, the boron concentration remains low due to dispersal throughout the RCS, so hot and cold leg injection is not essential).

Reactor coolant system pressure is used to differentiate between small and large break LOCA's. However, the delineation between small and large breaks does not need to be precise since there is a range of intermediate breaks for which either response will produce satisfactory results. The guidelines take this into account with the decisions to be made after eight hours.

The large break LOCA is characterized by:

a) A rapid loss of RCS pressure in 10 seconds to 3 minutes with equilibrium pressures below* 300 psia and, in the case of the largest breaks, the RCS pressure nearly equal to containment pressure.
b) Core cooling is provided for by large flow from the injection system due to low RCS pressure. The flow from the break provides sufficient heat removal. Simultaneous hot and cold leg injection is required to prevent possible boric acid accumulation in the core.

A general description of large break LOCA operations follows:

Initially, the plant is hot and pressurized. A large break LOCA results in a rapid loss of inventory and pressure. Low pressurizer pressure initiates a SIAS which automatically actuates the SIS. The reactor is tripped. Auxiliary feedwater is established to the steam generators. Steam dump is provided manually using atmospheric steam dump valves or turbine bypass valves. The major mechanism for heat removal is the flow from the SIS

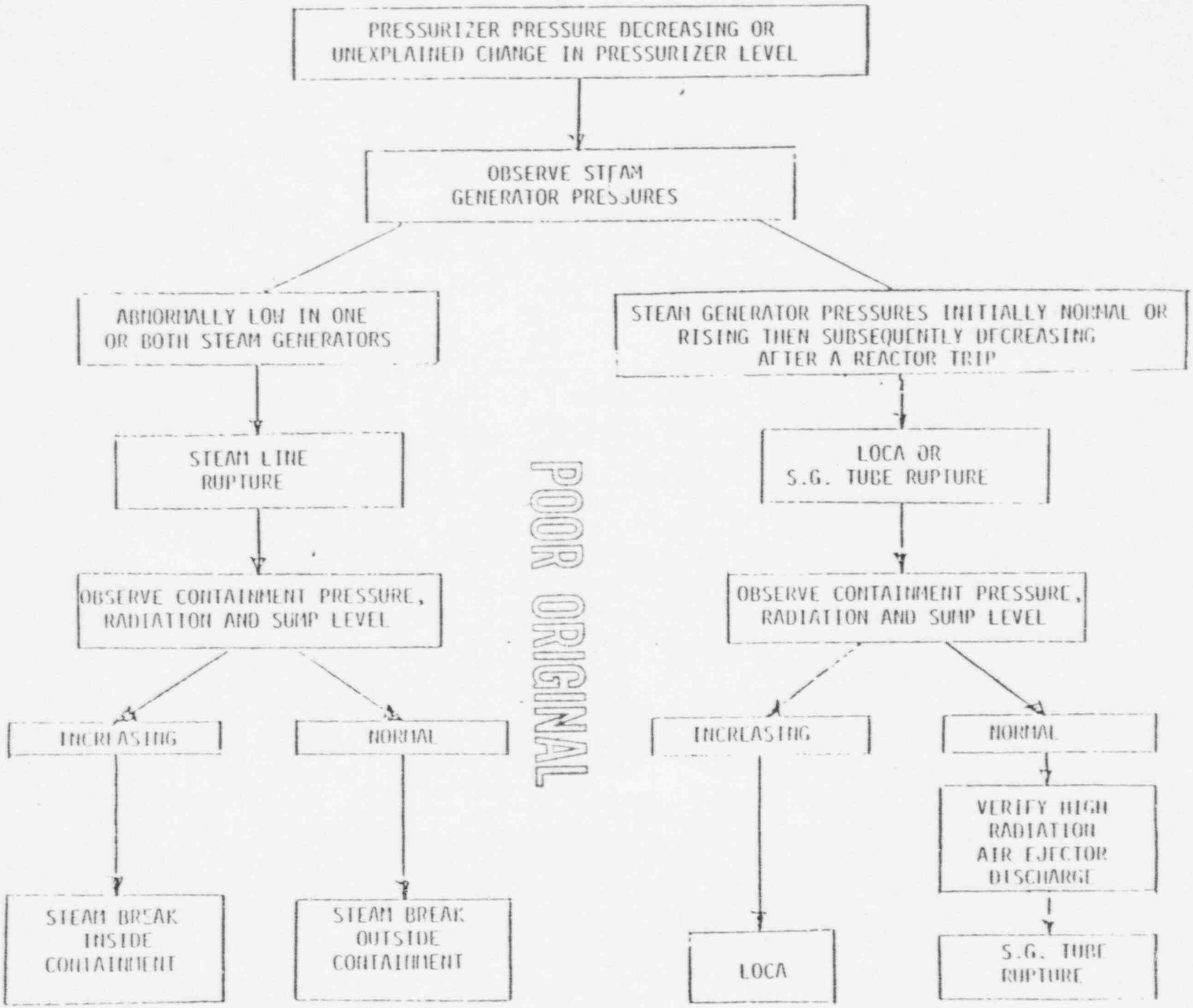
*This valve is typical, it may vary for specific designs.

through the core and out the break. Containment pressure may be high and containment isolation is likely. Containment spray may have been automatically activated.

The SIS is aligned to provide simultaneous hot and cold leg injection which is sufficient to cool the core and flush the reactor vessel indefinitely. For both large and small break LOCA's, continued monitoring of conditions in the RCS and performance of safety systems should be done. All available indications should be used to aid in diagnosing the event since the accident may cause irregularities in a particular instrument reading.

Regardless of the cause of actuation of a safety system, the automatic response should not be altered until it has been demonstrated that other systems and equipment are providing the functions that the safety system is intended to perform.

BREAK IDENTIFICATION



Guidelines for Operating Plants Following LOCA's

Symptoms

1. Reactor coolant system leak exceeds the capacity of the operable charging pumps.
2. A reactor trip may have occurred.
3. The Safety Injection System (SIS) may have automatically actuated.
4. Any one or more of the following indications or alarms may be present.
   a) Low pressurizer pressure
   b) High containment pressure or temperature
   c) High containment sump level
   d) High containment radiation
   e) High or low pressurizer level
   f) High quench tank level
   g) High quench tank temperature
   h) High quench tank pressure
   i) T_{av} decreasing or at saturation temperature for RCS pressure.

Immediate Actions

1. Trip the reactor if not already tripped and carry out standard post trip actions.
2. Initiate safety injection if it has not already been actuated by the safety injection actuation signal.
3. After an SIAS caused by low reactor coolant system pressure and after it has been verified that all rods have been fully inserted for 5 seconds, stop all operating reactor coolant pumps.
4. If main feedwater is not available, immediately establish or verify an auxiliary feedwater flow of *gpm.
5. If the containment isolation actuation signal (CIAS) is activated, ensure that the system has properly actuated.
6. Ensure that the systems receiving an SIAS are properly actuated and that CIAS is actuated.
7. After any SIAS, operate the SIS** until RCS hot and cold leg temperatures are at least 50^{o}F below saturation temperature for the RCS pressure and a pressurizer level is indicated, unless the cause of the SIAS has been verified to be an inadvertent actuation. If 50^{o}F subcooling cannot be maintained after the system has been stopped, the high pressure injection system must be restarted.

6. Attempt to locate and isolate the source of the leak. Possible leak locations include, but are not limited to the PORV's, the letdown line and sample lines.

Follow-Up Actions

1. Operate atmospheric steam dump valves (or turbine bypass valves if the condenser is available) to maintain or reduce plant temperature and reduce steam generator pressure below the steam generator relief valve setpoints. Begin a plant cooldown as soon as possible and in any case within 1 hour.
2. Manually align the safety injection and charging systems to provide flow to the RCS hot and cold legs* two hours after the LOCA**.
3. If the pressure and inventory control with the SIS cannot be established after* eight hours and RCS pressure is less than* 300 psig, continue the hot and cold leg injection.
4. If pressure and inventory control with the SIS are established after* eight hours and RCS pressure is greater than* 300 psig, conduct one of the following activities. The activities are listed in order of decreasing preference.

   a) RCS pressure above* 300 psig indicates that the system has refilled and subcooling has occurred. Verify this by checking the saturation pressure for the existing temperature. Realign the SIS for cold leg injection. Continue to maintain subcooling and reduce RCS pressure to the initiation pressure for shutdown cooling by reducing the flow delivered by the high pressure injection and charging pumps and by venting or isolating the safety injection tanks as necessary. While reducing pressure and after shutdown cooling is initiated, maintain RCS pressure with the charging pumps and/or the HPSI pumps to continue to maintain at least 50° subcooling, or

   b) Continue to remove decay heat using emergency feed and steam dump if adequate condensate is available and (a) cannot be implemented, or
   c) Open pressurizer power operated relief valves and align the SIS for cold leg injection if (a) or (b) cannot be implemented.

* This value is typical, it may vary for specific designs.

** Includes stopping charging pumps on some plants

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Precautions

1. Before restarting RCP's ensure that cooling water services to the pumps has been restored.

2. Pressurizer level may not always be a true indicator of RCS fluid inventory. Pressurizer steam space ruptures, reference leg failures, and reference leg flashing may cause indications which are contrary to true conditions.

3. All available indications should be used to aid in diagnosing the event since the accident may cause irregularities in a particular instrument reading. Critical parameters must be verified when one or more confirmatory indications are available.

4. When extablishing auxiliary feedwater flow to the steam generators, monitor primary system temperature and pressure to avoid exceeding a $100^{o}F$/hour cooldown rate.

5. Feedwater is normally provided to both steam generators. Isolation of a single steam generator is mandatory if a steam generator tube rupture is detected in that generator to prevent lifting of the safety valves or reseat them if they have lifted. This action will also reduce the amount of radioactivity released. For small breaks in the RCS where steam generators are important for heat removal one steam generator must be used for this purpose even if primary to secondary leaks are detected.

6. Continued lengthy operation of the containment spray may jeopardize the operation of equipment which would be desirable or necessary to mitigate the consequences of the event. Early consideration should be given to termination of spray operation. If the containment pressure has returned to below the actuation setpoint, the system may be stopped. The system should be realigned for automatic actuation.

7. Observe all available indications to determine conditions within the RCS. Use RCS hot leg temperature, RCS cold leg temperature, core exit thermocouple temperature, and RCS pressure to determine if the RCS is subcooled or saturated. An increase in temperature above the saturation temperature for the existing pressure is an indication of voiding in the RCS. A decrease in operating RCP motor current or erratic pump ΔP is also an indication of voiding. If this occurs the operator must ensure that the RCP's are turned off, the SIS is providing makeup to the RCS, and that the steam generators are removing heat from the RCS.

8. Monitor refueling water tank level to verify the shift from injection to recirculation. If a recirculation actuation signal (RAS) occurs, the operator must prevent the HPSI pumps from operating at less than minimum flow conditions. If all HPSI pumps and charging pumps are operating and the HPSI pumps are delivering less than 30 gpm per pump, turn off the charging pumps one at a time and then HPSI pumps one at a time until only one HPSI pump remains operating. This will ensure that minimum flow requirements will be met by the flow through the pump to the RCS for the smallest break size that results in a SIAS.

9. Monitor the auxiliary building radiation levels and sump levels after an RAS to attempt to detect leakage from the SIS. Even if leaks are detected at least one high pressure safety injection pump must remain in operation to provide flow to the RCS.

10. If there is a high radioactivity level in the reactor coolant system, circulation of this fluid in the SCS may result in high area radioactivity readings in the auxiliary building. The activity level of the RCS should be determined prior to initiating SCS flow.

11. Minimum Pressure - Temperature operating restrictions take precedence over requirements for operation of the high pressure injection or charging system to achieve 50^{o} subcooling during operation of the shutdown cooling system.

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