GE Nuclear Energy

175 Curtner Ave., San Jose, CA 95125

GE-NE-T43-00002-00-03-R01
Revision 1
DRF# T43-00002-00
Class I
August 1999

BWROG Position on the Use of Safety Relief Valves and Low Pressure Systems as Redundant Safe Shutdown Paths

Prepared by GE Nuclear Energy with the BWR Owners' Group Appendix R Committee

Prepared by:	J. L. Ribeiro	Date: 08/25/1999
Verified by:	B. P. Grim for - V. T. Shafer	Date: 08/31/1999
Approved by:	G. B. Stramback *Regulatory Services*	Date: 08/31/1999

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

TABLE OF CONTENTS

Executive Summary

This report provides a discussion on the BWR Owners' Group (BWROG) position regarding the use of Safety Relief Valves (SRVs) and low pressure systems (LPCI/CS) for safe shutdown. The BWROG's Position is as follows:

- The use of SRVs and Low Pressure Systems is an acceptable methodology for achieving redundant and/or alternative safe shutdown in accordance with the requirements of 10 CFR 50 Appendix R, Sections III.G. 1, 2 and 3. The distinction between the terms redundant and alternative safe shutdown as used in this document is described in Section 1.0 of this document.
- Based on the analytical and experimental justification referenced in this document, a short period of core uncovery for post-fire safe shutdown is acceptable. This condition will not result in any fuel cladding damage, rupture of the reactor coolant pressure boundary or rupture of the primary containment.

This report examines the acceptability of this position from the perspective of safety and risk significance, regulatory compliance and regulatory burden. The conclusions of this report are as follows:

- The risk, assessed in terms of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), associated with using SRVs and Low Pressure Systems as a redundant safe shutdown methodology is as low or lower than when using a high pressure system safe shutdown methodology.
- The use of SRVs and Low Pressure Systems in support of Appendix R Safe Shutdown is consistent with the original design basis for the GE BWR and, as such, it is a technically acceptable and safe means of achieving and maintaining either hot or cold shutdown. When used in the manner described in this report, there will be no fuel cladding damage, no rupture of the reactor coolant pressure boundary or no rupture of the primary containment.
- Acceptance of the use of SRVs and Low Pressure Systems for redundant safe shutdown, will avoid cost expenditures ranging from $0.2 to as much as $20.0 million for BWRs that currently use this approach for redundant safe shutdown.
- The BWROG Position on using SRVs and Low Pressure Systems for redundant safe shutdown is in full compliance and agreement with the regulatory requirements and guidance issued by the NRC relative to Appendix R Safe Shutdown.

1.0 INTRODUCTION

The purpose of this report is to explain and document the position of the GE BWROG on the use of Safety Relief Valves (SRVs) and low pressure systems (LPS) including low pressure coolant injection (LPCI) and/or core spray (CS), for satisfying the requirements of Appendix R Section III.G.1 and 2.

This report establishes a basis regarding the use of SRVs and LPS as a redundant shutdown methodology. Included in this report is a comprehensive review of this issue from the perspective of safety and risk significance, regulatory compliance and regulatory burden.

Based on the results of the reviews performed from these perspectives, an integrated analysis of the compiled facts will be performed and from this integrated analysis, conclusions will be drawn relative to the acceptability of the BWROG Position.

Section III.G.1 of Appendix R to 10 CFR 50 requires that one train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage and that, if repairs are required to systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station(s), that these required repairs be completed within 72 hours. Section III.G.2 of Appendix R requires specific fire protecting separation features for the redundant train selected for achieving post-fire safe shutdown in each fire area. Safe shutdown meeting the requirements of Sections III.G.1 and 2 of Appendix R is classified as redundant safe shutdown.

The systems available for use in achieving redundant post-fire safe shutdown for a BWR are summarized in GE Report, GE-NE-T43-00002-00-01-R01, "Original Safe Shutdown Paths for the BWR," August 1999. These systems were originally designed into the BWR without regard to Appendix R, and can function to achieve and/or maintain safe shutdown, including post-fire safe shutdown. The GE Report referenced above discusses various ways in which these systems may be combined to achieve and/or maintain post-fire safe shutdown by accomplishing the functions of reactivity control, reactor coolant make up, reactor vessel depressurization and reactor decay heat removal. Any combination of these systems with the capability to perform all of the shutdown functions listed above is considered to be an acceptable redundant post-fire safe shutdown path. One post-fire safe shutdown path must be assured to be free of fire damage and capable of performing its function in support of post-fire safe shutdown assuming a single fire in each plant fire area.

For those plant fire areas where none of the redundant systems can be protected as required by Section III.G.2 of Appendix R, Section III.G.3 requires that alternative post-fire safe shutdown capability which is independent of the fire area be provided. In addition, Section III.G.3 also requires that fixed suppression and detection be provided in those areas where alternative shutdown is used to achieve post-fire safe shutdown.

In this document, the specific regulatory requirements, guidance and information available will be reviewed to determine if SRVs and low pressure systems can be used as a redundant post-fire safe shutdown path. As discussed in this document, the NRC has already accepted the use of SRVs and low pressure systems in support of alternative post-fire safe shutdown.

Many BWRs have credited the use of SRVs and low pressure systems for redundant post-fire safe shutdown. The purpose of this report is to justify the acceptability of the positions taken by these individual BWR Licensee's. There is no intent that this paper will be used to justify the elimination or abandonment of existing plant fixed suppression and detection systems previously installed to meet the requirements of Section III.G.3 of Appendix R.

2.0 SUMMARY

The use of SRVs and LPS is an acceptable methodology for achieving redundant and alternative safe shutdown in accordance with the requirements of 10 CFR 50 Appendix R, Sections III.G. 1, 2 and 3.

Based on the analytical and experimental justification referenced in this document, a short period of core uncovery for post-fire safe shutdown is acceptable. This condition will not result in any fuel cladding damage, rupture of the reactor coolant pressure boundary or rupture of the primary containment.

3.0 COMPREHENSIVE REVIEW

3.1 REGULATORY REVIEW

The regulatory review is performed based on a hierarchy of regulatory documents which begins with the regulatory requirements contained in the code of federal regulations and the statements of consideration published at the time that Appendix R was promulgated. In addition, this review gives consideration to the information contained in NRC Generic Letters, NRC Information Notices and other pertinent documents and information, such as, SECY Documents, NRC Inspection Modules, internal NRC Memorandums and Licensee's plant specific commitments and the NRC SERs written to accept the license commitments. Although not all of this information is considered to be part of the regulatory requirements associated with this issue, the information can be used to understand the intent of the regulation where the specific intent has not been clearly addressed in the regulation.

This section of the report will examine each of the available regulatory documents, state its intended purpose relative to Appendix R compliance and explain any information from the document that is relevant to the topic of using SRVs and LPS in support of Appendix R Safe Shutdown. Verbatim wording extracted from the regulatory documents is italicized.

3.1.1 10 CFR 50 Appendix R and the Statements of Consideration

10 CFR 50 Appendix R Section III.G, provides the requirements for protecting structures, systems, components, cables and associated circuits required for achieving Appendix R Safe Shutdown.

Section III.G.1 provides the requirements for structures, systems and components and states the following:

III. G. Fire protection of safe shutdown capability.

1. Fire protection features shall be provided for structures, systems, and components important to safe shutdown. These features shall be capable of limiting fire damage so that:

a. One train of systems necessary to achieve and maintain hot shutdown conditions from either the control room or emergency control station(s) is free of fire damage; and

b. Systems necessary to achieve and maintain cold shutdown from either the control room or emergency control station(s) can be repaired within 72 hours.

There are no functional requirements specifically itemized for the structures, systems or components. The only performance goal identified is the requirement to initially achieve and maintain hot shutdown and to subsequently achieve cold shutdown once any required repairs have been completed.

Section III.G.1 establishes the requirement to ensure that adequate fire protection features exist to assure that one train of systems necessary to achieve and maintain hot shutdown is not impacted by the fire. Section III.G.2 specifies in detail those fire protection features that are adequate to ensure the protection of the hot shutdown train, including associated circuits. Section III.G.2 also introduces the term "redundant".

III.G.2. Except as provided for in paragraph G.3 of this section, where cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area outside of primary containment, one of the following means of ensuring that one of the redundant trains is free of fire damage shall be provided:

a. Separation of cables and equipment and associated non-safety circuits of redundant trains by a fire barrier having a 3-hour rating. Structural steel forming a part of or supporting such fire barriers shall be protected to provide fire resistance equivalent to that required of the barrier;

b. Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening *combustible or fire hazards. In addition, fire detectors and automatic fire suppression system shall be installed in the fire area; or*

c. *Enclosure of cable and equipment and associated non-safety circuits of one redundant train in a fire barrier having a 1-hour rating. In addition, fire detectors and an automatic fire suppression system shall be installed in the fire area;*

Inside non-inerted containments one of the fire protection means specified above or one of the following fire protection means shall be provided:

d. *Separation of cables and equipment and associated non-safety circuits of redundant trains by a horizontal distance of more than 20 feet with no intervening combustibles or fire hazards;*

e. *Installation of fire detectors and an automatic fire suppression system in the fire area; or*

f. *Separation of cables and equipment and associated non-safety circuits of redundant trains by a noncombustible radiant energy shield.*

Therefore, in order to comply with the regulatory requirements in Section III.G.1 and 2, it is necessary to: (1) provide fire protection features consistent with the requirements of Section III.G.2.a, b, or c within a fire area to ensure that structures, systems, components, cables and associated circuits for one train are capable of achieving and maintaining hot shutdown conditions; (2) assure that any repairs required to equipment necessary to achieve and maintain cold shutdown can be made within 72 hours.

These requirements, however, do not preclude a prompt transition to cold shutdown. In fact, a prompt transition to cold shutdown is a more desirable approach than maintaining hot shutdown conditions while cold shutdown repairs are performed. This position is supported by the NRC position provided in the Statements of Consideration for Appendix R (Ref. Federal Register #45FR76602 AT 50-SC-55, dated 9/1/82).

Part 50 Statements of Consideration (Comment Resolution for III.L) state the following: *"It is generally understood that cold shutdown is the ultimate safe shutdown condition and that, for each fire area, different means may be used and may be necessary to achieve cold shutdown."*

Therefore, demonstrating the ability to achieve and maintain cold shutdown following a postulated fire as described in Section III.G.2.a, b or c is the ultimate goal of Appendix R.

Section III.G.2 also makes provisions for the actions required in the event that the fire protection features described in Section III.G.2.a, b or c cannot be met[1]. In these cases, Section III.G.2 invokes the requirements of Section III.G.3. Section III.G.3 introduces the terms "Alternative" and "Dedicated" shutdown capability. Section III.G.3 also provides additional fire protection features that apply to those situations where alternative or dedicated shutdown capability is used to provide safe shutdown. Section III.G.3 reads as follows:

3. Alternative or dedicated shutdown capability and its associated circuits[2], independent of cables, systems or components in the areas, room or zone under consideration, shall be provided:

> *a. Where the protection of systems whose function is required for hot shutdown does not satisfy the requirement of paragraph G.2 of this section; or*
>
> *b. Where redundant trains of systems required for hot shutdown located in the same fire area may be subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems.*

In addition, fire detection and a fixed fire suppression system shall be installed in the area, room, or zone under consideration.

III.G.3 Footnote 2 - Alternative shutdown capability is provided by rerouting, relocating or modificating(sic) of existing systems; dedicated shutdown capability is provided by installing new structures and systems for the function of post-fire shutdown.

To satisfy the requirements of Section III.G.3 and use "Alternative" or "Dedicated" shutdown capability, the cables, systems or components comprising the "Alternative" or "Dedicated" shutdown capability must be independent of the area under consideration. Also by the court case referenced in IN 84-09 Attachment 1 Section VIII, Connecticut Light & Power vs. the NRC, 673 F2d. 525 (D. C. Cir.), cert. denied (1982), the requirements of Section III.L apply to the alternative safe shutdown option under Section III.G. This court case ruling provides clarification to the words contained in the Statements of Consideration for Appendix R, which reads as follows:

[1] By court ruling at the time of promulgation of Appendix R, licensee's required by 10 CFR 50.48 to meet the requirements of Appendix R may provide, for NRC approval, exemptions which demonstrate an equivalent level of fire protection to that required by Section III.G.2.a, b or c. By NRC Generic Letter 86-10, licensee's not required by 10 CFR 50.48 to meet the requirements of Appendix R and who have the "Standard License Condition" may demonstrate equivalency to the requirements of Section III.G.2.a, b or c using the 10 CFR 50.59 process.

Part 50 Statements of Consideration (Technical Basis for III.G) states: *Section III.L, "Alternative and Dedicated Shutdown Capability," discusses the technical basis for safe shutdown capability.*

The court case clarifies that these words apply to Section III.G.3.

Section III.L.1 provides requirements on the shutdown functions required for the systems selected for alternative shutdown. It also provides the acceptance criterion for the systems performing these functions.

L. Alternative and dedicated shutdown capability.

1. Alternative or dedicated shutdown capability provided for a specific fire area shall be able to (a) achieve and maintain subcritical reactivity conditions in the reactor; (b) maintain reactor coolant inventory; (c) achieve and maintain hot standby[3] conditions for a PWR (hot shutdown[3] for a BWR), (d) achieve cold shutdown conditions within 72 hours; and (e) maintain cold shutdown conditions thereafter. During the postfire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal a.c. power, and the fission product boundary integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of any primary coolant boundary, or rupture of the containment boundary.

[3] As defined in the Standard Technical Specifications

Section III.L.2 identifies the performance goals for the shutdown functions of <u>alternative shutdown systems</u> as:

2. The performance goals for the shutdown functions shall be:

a. The reactivity control function shall be capable of achieving and maintaining cold shutdown reactivity conditions.

b. The reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core for BWRs and be within the level indication in the pressurizer for PWRs.

c. The reactor heat removal function shall be capable of achieving and maintaining decay heat removal.

d. The process monitoring function shall be capable of providing direct readings of the process variables necessary to perform and control the above functions.

e. The supporting functions shall be capable of providing the process cooling, lubrication, etc., necessary to permit the operation of the equipment used for safe shutdown functions.

Therefore, "Alternative" shutdown capability meeting the requirements of Section III.G.3 must satisfy the requirements of Section III.L.

3.1.2 NRC Generic Letters

There are two NRC generic letters that provide information that is considered to be pertinent to this issue. NRC Generic Letter 86-10 was issued on April 24, 1986 in an effort to summarize the results of the Regional Workshops held in the Spring of 1984 regarding the implementation of NRC fire protection requirements at nuclear power plants. The guidance in this generic letter was intended to take precedence over prior guidance documents in the event of a conflict. NRC Generic Letter 81-12 was issued on February 20, 1981 as an information request intended to expedite the review process and reduce the number of requests for additional information associated with the NRC review of the alternative shutdown capability for plants licensed prior to January 1, 1979. Although this generic letter deals exclusively with a request for information related to "Alternative" shutdown capability, in the cover letter, a request was made that the results of the reassessment of the design features for meeting the requirements of Sections III.G, III.J and III.O be submitted.

NRC Generic Letter 86-10, Question 3.8.3 requests a clarification on the classification of redundant trains versus alternative shutdown. The question and the response are provided below.

3.8.3 Redundant Trains/Alternate Shutdown

QUESTION

Confusion exists as to what will be classified as an alternate shutdown system and thus what systems might be required to be protected by suppression and detection under Section III.G.3.b. For example, while we are relying upon the turbine building condensate system for a reactor building fire and the RHR system for a turbine building fire, would one system be considered the alternative to the other. If so, would suppression and detection be required for either or both systems under III.G.3.b? An explanation of alternative shutdown needs to be advanced for all licensees.

RESPONSE

If the system is being used to provide its design function, it generally is considered redundant. If the system is being used in lieu of the preferred system because the redundant components of the preferred system does not meet the separation criteria of Section III.G.2, the system is considered an alternative shutdown capability. Thus, for the example above, it appears that the condensate system is providing alternative shutdown capability in lieu of separating redundant components of the RHR System.

Fire detection and a fixed fire suppression system would be required in the area where separation of redundant components of the RHR system is not provided. However, in the event of a turbine building fire, the RHR system would be used for safe shutdown and is not considered an alternative capability. However, one train of the RHR system must be separated from the turbine building.

In the response, two relevant points are made. First, it is stated that a system is considered to be redundant when it is used to provide its design function. The RHR and condensate systems both have the design function of achieving and maintaining cold safe shutdown. The second is related to the introduction of the term "preferred system." Both the RHR and condensate systems are as-designed "preferred" systems for achieving and maintaining cold safe shutdown. "Preferred" is not defined in the response any further than to say that components of either division of the RHR system would be preferred to the condensate system.

NRC Generic Letter 86-10, Question 5.1.2 requests a clarification on the applicability of Section III.L to redundant trains and alternative shutdown. The question and the response are provided below.

5.1.2 Pre-Existing Alternative Shutdown Capability

QUESTION

Some licensees defined safe shutdown for purposes of analysis to Section III.G criteria as being composed of both the normal safe shutdown capability and the pre-existing redundant or remote safe shutdown capability which was previously installed as a part of the Appendix A process. This definition often took the form of two "safe shutdown trains" comprising (1) one of the two normal safe shutdown trains, and (2) a second safe shutdown train capability which was being provided by the pre-existing remote shutdown capability. This definitional process, which was undertaken by a number of licensees, makes a significant difference in the implementation of Appendix R. Under such a definition, does Section III.L criteria apply when the Commission did not call out Section III.L as a backfit?

RESPONSE

The definitional process mentioned considers an alternative shutdown capability provided under the Appendix A review as a redundant shutdown capability under the Appendix R review. This definitional process is incorrect. For the purpose of analysis to Section III.G.2 criteria, the safe shutdown capability is defined as one of the two normal safe shutdown trains. If the criteria of Section III.G.2 is not met, an alternative shutdown capability is required. The alternative shutdown capability may utilize the existing *remote shutdown capabilities and must meet the criteria of Sections III.G.3 and III.L of Appendix R. See also the response to 5.1.3.*

The response to this question provides information that clearly states that the requirements of Section III.G.3 and III.L apply to alternative shutdown. It can also be inferred from the response that the requirements of Section III.L do not apply to Section III.G.2.

NRC Generic Letter 81-12 was addressed to all Power Reactor Licenses with plants licensed prior to January 1, 1979 as indicated in the introduction.

Paragraph 50.48(b) of 10 CFR Part 50, which became effective on February 17,1981, requires all nuclear plants licensed to operate prior to January 1, 1979 to meet the requirements of Section III.G, III.J and III.O of Appendix R to 10 CFR Part 50 regardless of any previous approvals by the Nuclear Regulatory Commission (NRC) for alternative design features for those items.

In discussing "Alternative" shutdown capability, GL 81-12 provides an example list of systems generally provided for both hot and cold shutdown in a BWR along with the shutdown function each typically provides. The list provided in the generic letter for hot shutdown equipment is provided below.

6. BWR Equipment Generally Necessary For Hot Shutdown

(1) Reactivity Control

Reactor trip capability (scram)

(2) Reactor Coolant Makeup

Reactor coolant inventory makeup capability e.g., reactor core isolation cooling system (RCIC) or the high pressure coolant injection system (HPCI).

(3) Reactor Pressure Control and Decay Heat Removal

Depressurization system valves or safety relief valves for dump to the suppression pool. The residual heat removal system in steam condensing mode, and service water system may also be used for heat removal to the ultimate heat sink.

(4) Suppression Pool Cooling

Residual heat removal system (in suppression pool cooling mode) service water system to maintain hot shutdown.

(5) Process Monitoring

Process monitoring capability e.g., reactor vessel level and pressure and suppression pool temperature.

(6) *Support*

Support capability e.g., onsite power source (AC & DC) and their associated distribution systems to provide for the shutdown equipment.

This listing, however, in no way states or implies that the list is an exclusive list of equipment or systems necessary to achieve these functions and support the performance goal of safe shutdown. For example, in some post-fire situations, the path for decay heat removal through the main condenser may still be available. Under Section 6. (2) of Generic Letter 81-12, two examples (*e.g., reactor core isolation cooling system (RCIC) or high pressure coolant injection system (HPCI)*) of high pressure systems are identified. These systems, however, are not available in all plants operating before 1981 (e.g., BWR 1's, 2's, 3's) and hence, this list was not meant to be all inclusive and should not preclude the use of other systems to achieve the same function of providing reactor coolant makeup. Based on the information provided in NRC Generic Letter 81-12, the BWROG sees no requirement or basis for limiting the systems that may be used as redundant safe shutdown systems. Any system with an as-designed safe shutdown functional capability should be considered as redundant.

3.1.3 NRC Information Notices

NRC Information Notice 84-09 was issued on February 13, 1984 and revision 1 was issued on March 7, 1984 to provide information on the lessons learned from NRC inspections of Fire Protection Safe Shutdown Systems (10 CFR 50 Appendix R).

IN 84-09 Attachment 1 Section V discusses criteria which redundant safe shutdown systems and components must satisfy.

The NRC in IN 84-09 Attachment 1, Section V states the following criteria for selecting redundant safe shutdown systems and equipment.

The systems and equipment needed for post-fire safe shutdown are those systems necessary to perform the shutdown function defined in Section III.L of Appendix R. These functions are reactivity control, reactor coolant makeup, reactor heat removal, process monitoring, and associated support functions. The acceptance criterion for systems performing these functions is also defined in Section III. L:

During the post-fire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal a.c. power, and the fission product boundary integrity shall not be affected; i.e., there shall be no fuel clad damage, rupture of any primary coolant boundary, or rupture of the containment boundary.

These guidelines apply to the systems needed to satisfy both Section III.G and III.L of Appendix R.

The section of NRC IN 84-09 provides clarification that any combination of systems and components with the capability to satisfy the shutdown functions described above and meet the acceptance criteria described above is considered to be acceptable for redundant safe shutdown.

Two additional observations can be made relative to the information provided in Section V of NRC IN 84-09. The first is that the full scope of Appendix R, Section III.L does not apply to redundant shutdown governed by the requirements of Section III.G.1 and III.G.2. This can be inferred from the fact that the information provided reflects only a portion of the requirements of Appendix R, Section III.L and also from the fact that a later section of NRC IN 84-09 does invoke full compliance with Appendix R, Section III.L for III.G.3 compliance (see the next section of this report for more information). The second observation is that the requirement to maintain *the reactor coolant system process variables shall be maintained within those predicted for a loss of normal a.c. power* is not specifically defined. The specific process variables of concern and a clear definition of the condition described as "*a loss of normal a.c. power*" is not contained in the various regulations and guidance documents. As a result, the requirements for this condition are left to interpretation.

IN 84-09 Attachment 1 Section VIII discusses criteria which alternative shutdown systems must satisfy.

The NRC in IN 84-09, Section VIII states the following regarding the applicability of Section III.L requirements to Section III.G.3 Alternative Shutdown Systems.

VIII. Applicability of 10 CFR 50, Appendix R, Section III.L

Some of the inspected licensees had not considered Section III.L of Appendix R when attempting to meet Section III.G. The acceptance criteria for Section III.G.3 are listed in Section III.L. Although 10 CFR 50.48(b) does not specifically include Section III.L with Sections III.G., J, and O of Appendix R as a requirement applicable to all power reactors licensed prior to January 1, 1979, the Appendix, read as a whole, and the Court of Appeals decision on the Appendix, Connecticut Light and Power, et al. v. NRC, 673 F2d. 525 (D. C. Cir.), cert. denied (1982), does mean that Section III.L applies to the alternative safe shutdown option under Section III.G.

Therefore, Section III.L is applicable only to "alternative" shutdown systems as classified by Section III.G.3.

Based on a review of Section III.L, it is obvious that the Section III.L acceptance criteria deals strictly with those systems classified as "alternative" and is not intended for "redundant" systems.

3.1.4 Other Pertinent Documents and Information

3.1.4.1 NRC has approved use of SRVs with low pressure injection systems as redundant systems in existing BWRs

Attachment A lists the BWRs where the NRC has approved the use of SRVs with low pressure injection systems as a redundant safe shutdown methodology under the requirements of Section III.G.2. Attachment A provides a sense for the number of BWRs which would have to re-baseline their Appendix R analysis if this shutdown method was determined to be an unacceptable redundant shutdown path. As discussed later in this document, adjusting existing post-fire safe shutdown analysis to comply with this position would result in costs to licensees ranging from $0.2 to $20.0 million.

3.1.4.2 NRC has stated that the use of SRVs with low pressure injection systems complies with III.G.2 in Inspection Procedure 64100

NRC Inspection Procedure 64100, Post-Fire Safe Shutdown, Emergency Lighting and Oil Collection Capability at Operating and Near-Term Operating Reactor Facilities, states the following under Section 02.01, Section III.G.2, Redundant Train Safe Shutdown Capability; subsection a. Functional Requirements; paragraph 2.

The reactor coolant makeup function shall be capable of maintaining the level within the level indication of the pressurizer (or solid plant) for PWR's. For BWRs, the NRC has approved partial short-term core recovery using the automatic depressurization system (ADS) and low-pressure coolant injection system (LPCIS). Note that this option eliminates the need for the hot shutdown maintenance capability of Section III.G.1.a of Appendix R.

3.1.4.3 Rubenstein - Mattson 12/3/82 memo indicates that ADS/LPCI complies with III.L

The use of ADS and low pressure systems has been specifically reviewed and accepted by the NRC as a viable means of achieving safe shutdown for alternative shutdown conditions. The Rubenstein to Mattson Internal NRC Memo of 12/3/82 (Use of Automatic Depressurization System (ADS) and Low Pressure Coolant Injection (LPCI) to Meet Appendix R, Alternate Shutdown Goals) reads as follows:

In the course of performing Appendix R, safe shutdown reviews, the Auxiliary Systems Branch has noted that various boiling water reactor facilities propose the use of ADS in conjunction with LPCI as their proposed alternate shutdown method in order to achieve and maintain safe shutdown following a fire event in certain portions of the plant. For some cases this strategy will result in a short-term uncovering of the upper portion of the core during depressurization.

Analyses of similar events discussed in the reference give results which indicate that the uncovery time is short enough and the amount of fuel uncovered is small enough, that cladding integrity would not be threatened. The "lettered" requirements of Appendix R, however, would not be satisfied since Section III.L states:

1. *"...During the postfire shutdown, the reactor coolant system process variables shall be maintained within those predicted for a loss of normal AC power...", and*

2. *"The reactor coolant makeup function shall be capable of maintaining the reactor coolant level above the top of the core for BWRs..."*

The use of ADS and LPCI would not be the preferred means of maintaining reactor core cooling. Nevertheless, the use of ADS and LPCI is an approved and accepted means of achieving and maintaining safe shutdown conditions, and does comply with certain provisions of Section III.L of Appendix R regarding fission product boundary integrity.

Therefore, the use of SRVs and low pressure systems is considered to be an acceptable shutdown capability. There is no question as to whether the use of SRVs and low pressure systems is an acceptable alternative shutdown method. The temporary uncovery of the core has been established to be insignificant and should not have any bearing on the classification of whether these systems are redundant or alternative.

3.1.4.4 Preferred list of systems not defined in regulation but is defined in BWROG EPG

As stated in Section 3.1.2 of this report, NRC Generic Letter 86-10 in responding to Question 3.8.3 introduced the term "preferred." The term "preferred" is not clearly defined in either Generic Letter 86-10 or within other sections of the regulations related to fire protection. However, the term has appeared in NRC documents and correspondence.

The NRC Safety Evaluation of BWR OWNERS' GROUP - "Emergency Procedure Guidelines," NEDO-31331, Rev. 4, March 1987 Enclosure 2 page. 17-18 reads as follows:

"The following systems, designated as preferred injection systems, are used initially for RPV level control: Condensate/Feedwater, CRD, HPCI, RCIC (with suction from condensate storage tank defeating low RP pressure isolation interlocks and high suppression pool water level suction transfer logic if necessary), HPCS, RHR, LPCS (Vortex and NPSH limits are specified for pump protection)"

In this document the use of LPSs is included among the list of preferred injection systems.

3.2 SAFETY SIGNIFICANCE REVIEW

GE Report No. GE-NE-T43-00002-00-01-R01 "Original Safe Shutdown Paths For The BWR" identifies the systems and equipment originally designed into the GE Boiling Water Reactors (BWRs) that should be categorized as *original* (normal and redundant) and which can be used to achieve and maintain safe shutdown. The BWR design has always had in its basic design the philosophy of depressurization to low pressure systems. In fact, some of the original BWR designs did not even include a safety-related high pressure make up capability for accident conditions.

Section III.G.1 requires that one train of systems necessary to achieve and maintain hot shutdown conditions be free of fire damage. The use of SRVs for RPV pressure control coupled with the use of a low pressure system (i.e., Core Spray or RHR/LPCI) for RPV inventory make up meets this requirement. This is true because upon initial blowdown using SRVs the unit is in hot shutdown. Upon initial blowdown, the bulk reactor coolant temperature is around 212°F. At this point, hot shutdown conditions could be maintained, if desired, for an extended period of time by closing the SRVs and allowing the reactor to repressurize to a level below the shutoff pressure for the available low pressure system. In this mode of operation, injection flow would be throttled to maintain level within a specified, safe range and an SRV would be used to control pressure within a specified band. An operator might elect to maintain hot shutdown in this manner, if repairs were required to systems necessary to achieve and maintain cold shutdown. An operator, however, would not normally elect to do this if the ability to proceed directly to cold shutdown was available.

In using SRV/LPS, the potential does exist for a short term partial core uncovery depending on the time and reactor level conditions at which the reactor depressurization is initiated and the rate at which the depressurization is performed. This condition previously has been addressed from a safety significance perspective.

The Whitney to McKee Internal NRC Memo of 8/11/86, Enclosure 2, SECY 85-306 Meeting Minutes of 5/7/86 supports this point:

"Some attendees expressed concern over approved BWR ADS/LPCI post-fire safe shutdown configurations. Attendees were assured that fuel rod tests had been performed to assess the potential for core damage arising from short term partial core uncovery. DI contacted RES and developed the following information:

- *Dr. Robert Van Houten of the Fuel Systems Research Branch of the Division of Accident Evaluation, Office of Nuclear Regulatory Research (427-4463) is an authority in this area. He states that fuel rod testing has been conducted for many years at the National Reactor Universal at the Chalk River (U.S.) National Laboratory in Chalk River, Canada. Up to 32 bundled light water reactor fuel rods have been tested for short time periods in partial steam cooling mode with simulated 100% power history decay heat. The cladding partially oxidized but no fuel damage resulted."*

The conclusions of SECY-85-306 are consistent with the findings of NUREG-0562, which evaluated core uncovery events for a variety of reactor and fuel designs. Furthermore, analysis has demonstrated that the peak cladding temperature remains well within the design basis limits when using SRVs and low pressure systems. An example of the analysis performed for a typical BWR 4 is explained below.

Events with reactor and ECCS response similar to those experienced during an Appendix R postulated fire, have been analyzed as a part of the original plant licensing basis of each GE BWR. The following is a description of one such Design Basis Accident (DBA) and its relevance to the Appendix R postulated fire event when shutdown is accomplished using SRVs and low pressure systems to manually depressurize the reactor beginning when indicated water level reaches top of active fuel (TAF).

The event used to compare the reactor and ECCS System response with those associated with an Appendix R postulated fire is associated with a main steam line break outside of containment. In reviewing the information provided below, it should be noted that the initiating event, the main steam line break, has no bearing on the data from the portion of the scenario used in the comparison. This event was selected because of the similarities in shutdown methodology to an Appendix R Shutdown using SRVs and LPSs. The parameters of interest in making the comparisons, however, do not present themselves until after the initiating event has occurred, the MSIVs have fully closed, the scram has been completed and the reactor has stabilized. At this point, the vessel level is in a condition comparable to what it would be for an Appendix R postulated fire event where the reactor is scrammed and the MSIVs are closed.

In either event, decay heat from the fuel would repressurize the reactor to the setpoint of the SRV with the lowest setpoint. Because there is no high pressure make-up assumed in either event, level gradually "boils off" until it reaches the top of active fuel (TAF) in each event. Throughout this entire set of events, the reactor parameters respond comparably and the initiating event of the MSL break has no bearing on the performance of the reactor parameters subsequent to stabilization. This analysis demonstrates the response of the peak fuel cladding temperature in a BWR to conditions where level in the downcomer and the core region drop to below the top of active fuel. It is evident from a review of this analysis that no change in the fuel PCT occurs until level drops to below TAF.

3.2.1 Main Steam Line (MSL) Break Outside of Containment

The information provided below is based on GE Report GE-NE-187-22-0992 dated September 1993. The event is a main steam line break outside of containment. In conjunction with this event, HPCI is assumed to be lost due to a single failure and RCIC is assumed to be unavailable because it is not a safety-related system. With this DBE, the steam line break results in a closure of the MSIVs and a reactor scram. The MSIVs close on high steam flow. The reactor scrams from position switches indicating closure of the MSIVs (i.e. MSIVs ≤ 90% open). With the reactor in this condition, the reactor shutdown with the MSIVs closed, and with no high pressure sources available for injection, the remainder of the shutdown scenario is similar to a shutdown for an Appendix R postulated fire where SRVs and low pressure systems are used to achieve safe shutdown. By analyzing this scenario and highlighting the similarities and differences between this scenario and the Appendix R scenario, much can be learned about the reactor response, parameters and process variables during the Appendix R postulated fire event.

Attachment B to this report provides information on the reactor water level in the reactor during an analyzed main steam line break outside containment for a typical BWR 4. Attachment B also provides information on the reactor water level during a postulated

Appendix R fire scenario where shutdown is accomplished using SRVs and low pressure systems.

Attention will be focused first on the MSL break outside of containment event. Attachment B provides a plot of reactor water level versus time for the MSL break outside of containment event. Only the portion of the event that is pertinent to the Appendix R shutdown scenario is provided. Two individual plots are provided. The one with the higher water level is a plot of the coolant level inside of the core region. The other one is the water level in the downcomer region. Instrumentation readings based on the level in the downcomer region provide input to the ECCS automatic initiation logic. The level in the downcomer region is also the source of the level indication information provided to the operator in the Control Room.

For the analysis summarized in Attachment B, a water level of 30.5 feet represents the top of active fuel (TAF) and the level 1 setting for input into the ECCS automatic initiation logic. For the depressurization portion of the event, between 2109 and 2345 seconds, only five of the six available ADS/SRVs were assumed to open.

Attachment B provides information on the response of reactor level beginning approximately 1500 seconds after the start of the event. At this point, the reactor is isolated with no high pressure injection sources available. Reactor pressure is being controlled by the SRVs with the lowest setpoint valve opening and reclosing. The SRVs cycle repeatedly (assuming no operator action), gradually depleting the reactor inventory. At 1587 seconds, the reactor water level in the downcomer region reaches 30.5 feet, TAF and the level 1 trip. The ADS logic requires both a high drywell pressure and a low water level signal to initiate the ADS 102 second timer (note: this is a nominal delay setting, the upper limit for this setting is equal to 120 seconds). The steamline break outside of containment does not cause a high drywell pressure. Consequently, for ADS to initiate at this point would require manual initiation by the Control Room operator. Within the ADS logic, however, there is a bypass for the high drywell pressure based on an extended low water level signal. Typically, the low water level signal will bypass the high drywell pressure permissive if it exists for 7.0 minutes (note: this setting is a plant unique permissive setting).

Therefore, even though the low water level signal is reached at 1587 seconds, ADS does not initiate until 2109 seconds, 522 seconds later. The 522 second delay is a result of the 7.0 minute, 420 second, bypass timer and the ADS timer, 102 seconds. The 7 minute timer setting varies somewhat from plant to plant based on a calculation to avoid automatic ADS during a postulated transient with failure of scram (ATWS).

During this 522 second time, it can be noted that the level in the downcomer region drops below TAF while the two-phase coolant level in the core region stays above TAF. It can also be observed for this same time interval that the peak cladding temperature of the fuel does not change, but rather holds steady at 589^{0}F. This observation attests to the fact that the fuel temperature is affected by the level of the coolant in the core region and is unaffected by the water level in the downcomer region.

At 2109 seconds into the event, ADS initiates. The initiation of ADS results in the following: (1) a level swell in both the downcomer and core regions of the reactor; (2) a decrease in level as the open SRVs depressurize the reactor and deplete the reactor inventory; (3) a reduction in the fuel cladding temperature as the fuel temperature follows the saturation curve.

It should be noted that the decrease in fuel cladding temperature continues until 2246 seconds. At this point, the coolant level in the core region drops below TAF and this causes a gradual increase in peak fuel cladding temperature for approximately 2 minutes (127 seconds) until level in the core region is again restored to above TAF when core spray injection is capable of raising level to this point. Even with an uncovering of a portion of the core for approximately 2 minutes, the peak fuel cladding temperature reached a value of only 689°F, as compared to the licensing basis peak cladding temperature for this BWR of 1510°F and the ECCS Fuel Peak Cladding Temperature Limit of 2200°F. This portion of the timeline again emphasizes the fact that the fuel cladding temperature is affected by the level of the coolant in the core region and is unaffected by the water level in the downcomer region.

3.2.2 Appendix R Scenario

For the Appendix R scenario in which no high pressure sources of injection were available and level was dropping towards TAF, the Emergency Operating Procedures would instruct the Control Room operator to start systems capable of injecting at low pressure and to rapidly depressurize the reactor when level reaches TAF. Therefore, in the Appendix R scenario, where only low pressure systems were available, these systems would be lined up for injection with their respective pumps running prior to level reaching TAF. When level reaches TAF, the operator would initiate rapid depressurization of the reactor and as soon as the reactor pressure reached the shutoff head for the available low pressure system, injection would begin and level recovery would rapidly follow.

By using the bolded portions of the coolant level and water level plots from the MSL break event shown in Attachment B and moving them back to the point in the scenario where indicated downcomer level reaches TAF, the reactor level conditions for an Appendix R postulated fire scenario can be predicted. The time frame required to reach the shutoff head for the low pressure systems will be shorter in this Appendix R scenario because the number of SRVs used is greater. The number of SRVs used is plant specific and the minimum number of SRVs selected for a specific plant is in part based on plant specific calculations that demonstrate acceptable PCT values below the level that would result in any fuel cladding damage (either EPG calcs or Fire Event-Specific calcs). For the specific BWR 4 being addressed in this evaluation, the number of SRVs used is six (6). The shorter time frame required to reach the shutoff head for the low pressure system is reflected in the portion of the curves that was transposed. Also, because the low pressure systems will be running prior to rapidly depressurizing, reactor vessel injection will begin immediately upon the reactor pressure reaching the low pressure system shutoff head. This also is reflected in the portion of the curve transposed.

For this Appendix R scenario, it can be seen that the level of the two-phase reactor coolant in the core region remains above TAF. Based on the response of the fuel in the MSL break event, it also is clear that the peak cladding temperature will hold steady at 589°F when the level of the coolant in the core region remains above TAF. Finally, based on the response of the fuel in the MSL break event, it is clear that this temperature of 589°F will be the highest fuel cladding temperature throughout the scenario, because the fuel cladding temperature will follow the saturation curve and gradually decrease as reactor pressure decreases.

3.3 RISK INSIGHT

3.3.1 Background for Risk Comparison

The risk significance of using SRVs and low pressure systems will be evaluated in this section in terms of both Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). The evaluation will assess the risk significance of using SRVs and low pressure systems in a relative manner by comparing this shutdown methodology with a high pressure shutdown methodology. NRC Generic Letter 81-12, in Sections 6 and 7, endorses the use of the high pressure system shutdown system methodologies. The comparisons will be made in a qualitative sense to demonstrate, as a minimum, an equivalency to the accepted high pressure shutdown methodology.

Figure 1 illustrates the comparison being made in this section.

The assessment performed in this section focuses on a typical BWR 4 as an example. Although most aspects of the discussion provided are true for all BWRs, the discussions related to specific systems could vary from one BWR to another. These variations, however, are not judged to alter the overall conclusions of this document.

For a typical BWR 4, numerous methodologies are available for achieving and maintaining safe shutdown in the event of an Appendix R fire. Figure 1 depicts two safe shutdown methodologies, one using high pressure systems and one using low pressure systems, to achieve safe shutdown in a typical BWR 4. With either of these shutdown methodologies, the Scram function provided by the Control Rod Drive (CRD) System would be required to accomplish the reactivity control function.

Figure 1 shows two Division II safe shutdown methodologies:

(1) A High Pressure Shutdown Methodology consistent with that described in NRC Generic Letter 81-12 and comprised of HPCI, 1-2 SRV, RHR Shutdown Cooling (SDC) and the necessary Division II support systems [RHR Suppression Pool Cooling (SPC), RHRSW, Emergency Service Water (ESW) and the Electrical Distribution System (EDS)]. For this shutdown methodology, HPCI and 1 SRV are required to maintain hot shutdown in accordance with Appendix R Section III.G.1.a. RHR SDC is required for achieving cold shutdown. In accordance with Appendix R Section III.G.1.b, cold shutdown repairs will be made to the RHR SDC Suction

Valves and the RHR SDC Discharge Valves. These repairs must be completed within the 72-hour time period prescribed by Appendix R.

(2) A Low Pressure Shutdown Methodology comprised of ADS/SRVs, RHR Low Pressure Coolant Injection (LPCI), RHR SDC and the necessary Division II support systems [RHR Suppression Pool Cooling (SPC), RHRSW, Emergency Service Water (ESW) and the Electrical Distribution System (EDS)]. For this shutdown methodology, LPCI and ADS/SRVs are required to maintain hot shutdown in accordance with Appendix R Section III.G.1.a. RHR SDC is required for achieving cold shutdown. In accordance with Appendix R Section III.G.1.b, cold shutdown repairs will be made to the RHR SDC Suction Valves. Repairs will not be made to the RHR SDC Discharge Valves, because these valves also are the LPCI Injection Valves. Since LPCI is required for hot shutdown, Appendix R requirements will not allow these valves to be repaired. Those repairs that are allowed must be completed within the 72-hour time period prescribed by Appendix R.

The systems used in each of these methodologies are identical, except that the High Pressure Methodology uses HPCI and 1-2 SRV, while the Low Pressure Methodology uses LPCI/CS and ADS/SRVs. These systems, as used in the shutdown methodologies described above, are related to the reactor coolant make up function. The decay heat removal function for each of these methodologies is accomplished initially by RHR in the Suppression Pool Cooling Mode and later by RHR in the Shutdown Cooling Mode.

Appendix R requires that one hot shutdown path be available (i.e., free of fire damage) for achieving and maintaining safe shutdown assuming a single fire in any fire area. Therefore, referring to the methodologies depicted in Figure 1, in a Division II Safe Shutdown Fire Area where most of the Division I equipment would be affected by the fire, either the High Pressure or the Low Pressure Methodology would need to be assured to be unaffected by the fire. Where circuits for the selected shutdown methodology were identified to be located in the fire area of concern, these circuits would need to be protected from the effects of fire.

Section III.G.2 of Appendix R provides requirements for protection of the Redundant Safe Shutdown Train within each fire area. Section III.G.2 requires that the Redundant train be protected with either a 3-hour fire barrier, a 1-hour fire barrier when automatic suppression or detection is provided throughout the fire area or 20 foot separation distance containing no intervening combustibles with automatic suppression and detection. These requirements apply equally to the High Pressure and the Low Pressure Methodologies. Therefore, the level of fire protection provided for either shutdown methodology would be the same.

With respect to the fire protection features used in Nuclear Power Plant design and construction, Appendix A to Branch Technical Position 9.5.1 provides general guidance that safety-related equipment be isolated from unacceptable fire hazards and that fire protection features be provided in the locations where fire hazards are present. Because each of these shutdown methodologies relies upon safety-related equipment, the criteria for preventing and mitigating the effects of plant fires apply equally to each. Therefore, the potential for a fire effecting equipment for one or the other of these methodologies is the same.

Based on the discussion provided so far, the following conclusions can be drawn:

(1) Because the criteria used in the design and construction of the general plant fire protection features are not a function of shutdown methodology, neither the potential for a fire to occur nor the potential for fire spread is affected by the shutdown methodology selected.

(2) Because the requirements for protection of redundant safe shutdown circuits within a fire area are the same for each methodology, the potential for damage to these redundant safe shutdown circuits is the same.

(3) Because the decay heat removal function for each methodology is accomplished by the same systems performing the same functions within the bounds of their original plant and system design basis, there is no difference between these two methodologies in the potential for the rupture of the primary containment.

(4) Because both methods use the same equipment for decay heat removal, the change in LERF is directly proportional to a change in CDF.

(5) To assess the potential for a change in CDF, the systems whose capabilities must be compared are HPCI with a single SRV for the High Pressure Methodology and LPCI/CS and 1-2 ADS/SRVs for the Low Pressure Methodology. The primary safe shutdown function performed by these systems in their respective shutdown methodologies is reactor coolant make up. This capability relates directly to the ability to maintain reactor water level. For a BWR, reactor core coverage guarantees that there will be no fuel cladding damage. Significant and sustained fuel uncovery is necessary for fuel cladding damage to result. Fuel cladding damage is necessary for a radiological release.

3.3.2 Risk Comparison

Using the background provided above, this section provides a qualitative comparison of the potential for an increase in either Core Damage Frequency (CDF) or Large Early Release Frequency (LERF), when using either methodology. This comparison does not determine the actual CDF or LERF values for the Low Pressure Methodology. Rather the approach qualitatively demonstrates that the Low Pressure Methodology has no more potential for damaging the core or having a radioactive release than the already accepted High Pressure Methodology. By doing this, it is demonstrated that the potential for CDF or LERF for the Low Pressure Methodology is as low or lower than that for the High Pressure Methodology.

In assessing the risk associated with each of these methodologies, a comparison of system capabilities in terms of the ability to accomplish the required shutdown function of reactor coolant make up provides significant insight. This capability relates directly to the potential for an increase in CDF. If there is no potential for an increase in CDF when using the Low Pressure Methodology, then there also will be no potential for an increase in LERF.

A qualitative comparison of system reliability, system availability and human error potential when using these systems provides useful insights about the capability of each of these systems. Industry data provided in WASH 1400, supplemented by a consensus understanding of the data available from recent PRAs and IPEs and plant problems with the potential to impact plant safe shutdown leads to the following conclusions.

(1) The reliability of high pressure systems such as HPCI and RCIC is generally lower (i.e., higher failure probability) than for ADS/SRVs. For purposes of the comparison being performed in this document, it will be conservatively assumed that there is no difference.

(2) In various high pressure scenarios in PRAs and IPEs, human error contributors have had significance in terms of maintaining reactor water level. In the low pressure scenarios, these human error contributors have not proven to be significant other than assuring that the operator depressurizes the reactor at the proper time. Because in our scenario, however, manual operator actions are necessary to accomplish both HPCI initiation for the high pressure methodology and SRV initiation for the low pressure methodology, and, in either case, the time available to perform the action is the same, it is assumed that there is no difference.

(3) High pressure systems such as HPCI and RCIC have lower availabilities than the ADS/SRVs. For purposes of the comparison being performed in this document, it will be conservatively assumed that there is no difference.

Although the reactor coolant level could be lower when using the low pressure shutdown methodology, the analytical and experimental information provided in this document demonstrates that this condition does not affect the fuel cladding. Based on this and the information provided above, the CDF associated with the use of the low pressure shutdown methodology is as low or lower than the CDF associated with the high pressure shutdown methodology. Because the CDF for the low pressure shutdown methodology is as low as or lower than that for the high pressure methodology, it follows that the LERF for the low pressure shutdown methodology also is as low or lower.

In conclusion, there is no increased risk associated with using a low pressure shutdown methodology for redundant post-fire safe shutdown. The table below summarizes the capabilities important for shutdown and compares the differences between these two shutdown methodologies.

Shutdown Capability	Methodology Comparison
Fire Potential where used	No difference
Fire Spread Potential where used	No difference
Fire Mitigation Capability where used	No difference
Fire Damage Potential	No difference
Reactivity Control Capability	No difference
Reactor Coolant Capability	LPSs have higher volume
Decay Heat Removal Capability	No difference
System Reliability	LPSs more reliable, but assumed equal
Human Error Potential	No difference
System Availability	LPSs better availability, but assumed equal
Capability to protect Core	No difference
CDF	**LPSs, as a minimum, equal to HPSs**
LERF	**LPSs, as a minimum, equal to HPSs**

3.4 REGULATORY BURDEN

The regulatory burden associated with classifying the use of SRVs and low pressure systems strictly as an "Alternative" shutdown methodology and not allowing it to be used for redundant safe shutdown will vary from plant to plant depending on how extensively this methodology is used in the current safe shutdown analysis. Attachment A provides an indication of the number of BWRs that currently make use these systems as a "Redundant" safe shutdown methodology.

For a plant that uses this shutdown methodology in selected fire areas, the cost associated with altering the safe shutdown methodology on a limited basis would be less than $1,000,000.00. For example, one plant listed in Attachment A would require modifications costing over $750,000.00 in order to comply with the consequences of classifying this method as alternative shutdown. Another plant would require spending an estimated $200K simply to install a disconnect switch and reroute some cables.

For a plant that uses this shutdown methodology extensively throughout its safe shutdown methodology, the cost could be an order of magnitude greater, $10,000,000.00 to $20,000,000.00. For these plants, classifying the use of SRVs and low pressure systems as an "Alternative" shutdown methodology would require a complete revision to the current safe shutdown analysis. Because these plants have relied upon SRVs and LPS as a redundant safe shutdown methodology, these systems as currently designed would not be independent of the area, room or zone under consideration, and, as such, they would not satisfy the requirements of Appendix R, Section III.G.3. The net result of this condition would be that the current safe shutdown path would need to be abandoned, and the safe shutdown analysis would need to be re-performed to demonstrate the availability of a high pressure safe shutdown path. This effort would undoubtedly require a significant amount of plant circuit modifications and/or raceway protection.

Endorsing the use of SRVs and low pressure systems as an acceptable redundant shutdown method would eliminate the need for the significant expenditures summarized below:

Extent of Usage	Cost to Comply
Single Circuit Requiring Isolation	$0.2 million
Limited Fire Area	$0.75 million
Throughout Plant	$10.0 – $20.0 million

4.0 INTEGRATED ANALYSIS OF THE FACTS

A review of the facts provided under Section 3.1, Regulatory Review, supports the conclusion that it is acceptable to use SRVs and LPS as a redundant safe shutdown methodology under the requirements of Appendix R, Section III.G.1 and III.G.2. Although 10 CFR 50 Appendix R, Section III.G.1 and III.G.2 do not provide a clear basis for this conclusion, they do not provide any information that would lead to a conclusion that this shutdown methodology is unacceptable. The Connecticut Light & Power Court Case, the response to Question 5.1.2 in Generic Letter 86-10 and the distinctions made in IN 84-09, Attachment 1, Section V and VIII make it clear that III.L requirements apply only to Alternative shutdown systems under the requirements of Section III.G.3. Section III.G.3 to Appendix R clearly states that an "alternative" shutdown system is provided only when the requirements of Section III.G.2 cannot be satisfied by systems already in the plant design, and that when an "alternative" system is provided, it must be independent of the cables, systems or components in the areas, room or zone under consideration.

This requirement makes it clear that safe shutdown methodologies are classified as "alternative" based on the fire protection features of the area under consideration, and not on the basis of the systems which comprise the shutdown methodology. The response to Question 3.8.3 in Generic Letter 86-10 and the NRC's SER on Revision 4 to the EPGs state that low pressure systems are preferred systems for reactor inventory control. Furthermore, the December 3, 1982 Rubenstein – Mattson Memo states that the use of ADS and low pressure systems is considered to be an acceptable shutdown capability and the NRC Inspection Procedure 64100 for Post-Fire Safe Shutdown clearly states that the use of ADS and LPCIS is approved for III.G.2 shutdown.

When this information is combined with the information compiled under the safety significance review (Section 3.2), the risk significance review (Section 3.3) and the regulatory burden review (Section 3.4) of this report, the conclusion that it is acceptable to use SRVs and LPS as a redundant safe shutdown methodology under the requirements of Appendix R, Sections III.G.1 and III.G.2 is further solidified. Section 3.2 clearly concludes that the use of SRVs and LPS to rapidly depressurize the reactor and inject water to cool the core is not only fully consistent with the original design basis for the BWR, but also a shutdown methodology that provides for full protection of the core with no adverse safety consequences. Section 3.3 concludes that there is no increase in risk associated with the use of the SRV and LPS shutdown methodology for redundant post-fire safe shutdown. Section 3.4 concludes that significant cost expenditures would be required to change licensees safe shutdown analysis and plant features if SRVs and LPS were not allowed to be used in support of redundant post-fire safe shutdown.

In summary, it is concluded that using SRVs and LPS for redundant post-fire safe shutdown is acceptable from each perspective reviewed and that the failure to allow its use will result in significant and unnecessary costs to licensees.

5.0 CONCLUSIONS

This report examines the acceptability of the BWROG's evaluation on the use of Safety Relief Valves (SRVs) and low pressure systems (LPCI/CS) as redundant shutdown systems stated in Section 3.0 of the report from the perspective of safety/risk significance, regulatory compliance and regulatory burden. The conclusions of this evaluation are as follows:

- The risk, assessed in terms of Core Damage Frequency (CDF) and Large Early Release Frequency (LERF), associated with using SRVs and Low Pressure Systems as a redundant safe shutdown methodology is as low or lower than when using a high pressure system safe shutdown methodology.
- The use of Safety Relief Valves and Low Pressure Systems in support of Appendix R Safe Shutdown is part of the original design basis for the GE BWR, and, as such, it is a technically acceptable and safe means of achieving and maintaining redundant or alternative post-fire safe shutdown. When used in the manner described in this report, there is no fuel cladding damage, rupture of the reactor coolant pressure boundary or rupture of the primary containment. Low pressure shutdown methodologies using fewer numbers of SRVs and different low pressure make up capability (e.g., Core Spray) also are acceptable as long as these methodologies result in no fuel cladding damage, rupture of the reactor coolant pressure boundary or rupture of the primary containment.

- The classification of the SRVs and Low Pressure Systems shutdown methodology as an Alternative shutdown method would result in a significant and undue regulatory burden to the industry. This regulatory burden would result in significant expenditures for BWR licensees.
- The results of the BWROG evaluation is in full compliance and agreement with the regulatory requirements and guidance issued by the NRC relative to Appendix R Safe Shutdown.
- This report supports the acceptability of the positions taken by individual BWR Licensee's that the use of SRVs and low pressure systems is an acceptable redundant post-fire safe shutdown methodology. While individual licensees can use this information as they see fit, the BWROG does not intend to justify the elimination or abandonment of existing plant fixed suppression and detection systems previously installed to meet the requirements of Section III.G.3 of Appendix R.

6.0 REFERENCES

The following references are used to support the discussions in this report:

1. 10 CFR 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," Sections III.G, J, L, and O.
2. "Statements of Consideration for Appendix R" (Ref. Federal Register #45FR76602 AT 50-SC-55, dated 9/1/82).
3. Generic Letter 81-12, "Fire Protection Rule," February 20, 1981.
4. Generic Letter 86-10, "Implementation of Fire Protection Requirements," April 24, 1986.
5. IE Information Notice 84-09, "Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems," February 13, 1984. Revision 1 issued March 7, 1984.
6. Rubenstein to Mattson Internal NRC Memo of 12/3/82, "Use of Automatic Depressurization System (ADS) and Low Pressure Coolant Injection (LPCI) to Meet Appendix R, Alternate Shutdown Goals."
7. GE-NE-187-22-0992, NEDC-32281P, September 1993, DRF A00-04159, "SAFER/GESTR-LOCA Analysis Basis Documentation for Susquehanna Steam Electric Station Units 1 and 2."
8. GE-NE-T43-00002-00-01-R01, "Original Safe Shutdown Paths For The BWR," August 1999.
9. Whitney to McKee Internal NRC Memo of 8/11/86, Enclosure 2, SECY 85-306 Meeting Minutes of 5/7/86.

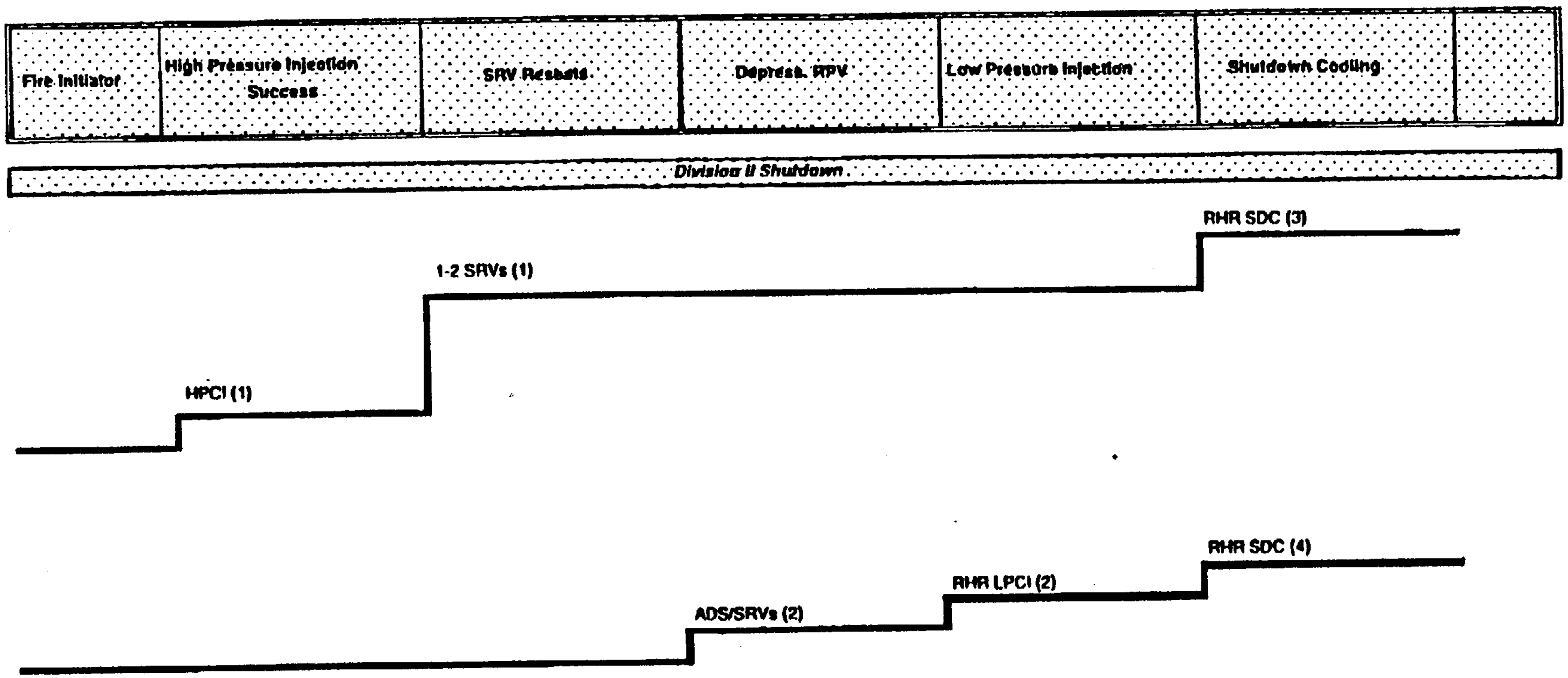
10. NRC Safety Evaluation of BWR OWNER'S GROUP - 'Emergency Procedure Guidelines," NEDO-31331, Rev. 4, March 1987.

11. NRC Inspection Procedure 64100, "Post-Fire Safe Shutdown, Emergency Lighting and Oil Collection Capability at Operating and Near-Term Operating Reactor Facilities," 03/16/87.

12. NUREG-0562 - "Fuel Rod Failure as a Consequence of Departure for Nucleate Boiling or Dryout," Robert Van Houten, June 1979.

13. WASH 1400, Reactor Safety Study, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Appendix II, October 1975.

Figure 1 - Representative Division II Appendix R Safe Shutdown Path



Notes

(1) HPCI and 1 SRV are required to maintain hot shutdown.

(2) LPCI and ADS/SRVs are required to maintain hot shutdown.

(3) Per Appendix R, Section III.G.1.b the RHR SDC Suction and Discharge Valves will be repaired within the 72 hour period allowed.

(4) Per Appendix R, Section III.G.1.b the RHR SDC Suction Valves will be repaired within the 72 hour period allowed. Cold shutdown also could be achieved using ADS/SRVs + LPCI/CS + RHR heat exchanger.

Attachment A

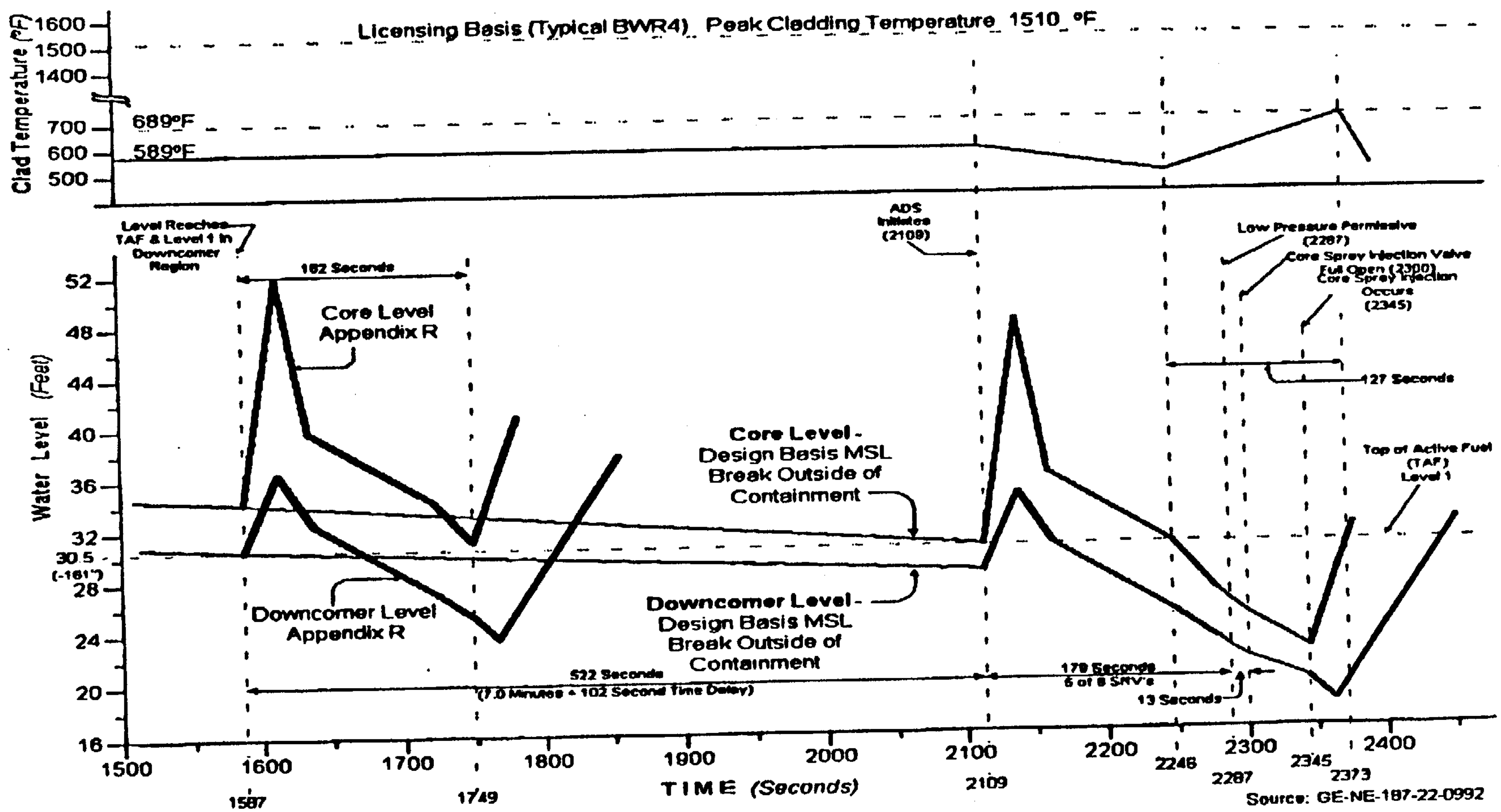
Summary on BWR Licensee's Use of ADS/SRV & Low Pressure Systems

Plant	Use of Low Pressure Systems	Used in III.G.2 Fire Areas	Accepted* by NRC
A	Yes	Yes	Yes
B	Yes	Yes	Yes
C	Yes	Yes	Yes
D	Yes	Yes	Yes
E	Yes	Yes	Yes
F	Yes	Yes	Yes
G	Yes	Yes	Yes
H	Yes	Yes	Yes
I	Yes	Yes	Yes
J	Yes	Yes	Yes
K	Yes	Yes	Yes
L	Yes	Yes	Yes
M	Yes	Yes	Yes
N	Yes	Yes	Yes

* Accepted by NRC could be in the form of an Safety Evaluation Report, NRC Inspection or approval of Safe Shutdown methodology.

Attachment B

Summary Plot of Peak Fuel Cladding Temperature & Water/Coolant Level versus time when using SRVs & Low Pressure Systems



GE Nuclear Energy

175 Curtner Ave., San Jose, CA 95125

GE-NE-T43-00002-00-01-R01
Revision 1
DRF T43-00002-00
Class I
August 1999

Original Safe Shutdown Paths For The BWR

Prepared by: Kurt T. Schaefer
Eugene C. Eckert
Daniel C. Pappone

Verified by: Britton P. Grim
Roger T. Earle
Donald C. Rennels

Approved by: George B. Stramback
Regulatory Services

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

The only undertakings of the General Electric Company (GE) respecting information in this document are contained in the contract between the BWR Owners Group (BWROG) and GE, as amended to the date of transmittal of this document, and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than the BWROG, or for any purpose other than that for which it is intended is not authorized. With respect to any unauthorized use, GE makes no representation or warranty, express or implied, and assumes no liability as to the completeness, accuracy or usefulness of the information contained in this document, or that its use may not infringe privately owned rights.

Summary

This report addresses the systems and equipment originally designed into the GE Boiling Water Reactors (BWRs) in the 1960's and 1970's, that can be used to achieve and maintain safe shutdown per Section III.G.1 of 10CFR 50, App. R. Although not specifically described throughout this document, these systems can also be used to satisfy the safe shutdown requirements of NRC Branch Technical Position 9.5.1 or Section 9.5.1 of the NRC Standard Review Plan. Based on guidance in USNRC Generic Letter 86-10 (Reference 1), the systems and equipment addressed herein generically represent the *original* (normal and redundant) safe shutdown paths.

This report includes some plant-specific guidance, and a general process for determining which plant-specific equipment and systems should be categorized as *original* (normal and redundant) shutdown capabilities. Due to the diversity of safe shutdown capability contained in the original systems provided for the GE BWR, it is possible that paths different from those described in this report, can be used to satisfy all of the requirements of Appendix R. On a plant-specific basis, the design capability of the systems described in this report can be used in different combinations to satisfy all of the requirements of Appendix R. These different paths would, in most cases, be a by-product of the specific fire damage in a fire area and of the repair schemes selected by the individual licensees. As such, this report does not generically describe all paths that could be developed for achieving and maintaining safe shutdown for all BWRs.

Table of Contents

Tables

Figures

1. Introduction/Background

To properly apply 10 CFR 50 Appendix R, the *original* safe shutdown paths for the BWR need to be reviewed. The *original* safe shutdown paths represent the normal and redundant shutdown paths, as originally designed into the BWR.

This report addresses the safe shutdown paths originally designed into the BWR by GE, prior to 10 CFR 50, Appendix R. Although some BWRs may have as their licensing basis a requirement only to achieve hot shutdown, these *original* safe shutdown paths, in addition to accomplishing this function, are expected to also be able to achieve cold shutdown in less than 72 hours.

Included are some of the typical differences between BWR product lines relative to their normal safe shutdown paths, while emphasizing the fact that all BWR product lines always have had in their basic design the philosophy of depressurization to low pressure systems.

Enclosure 1, Section 2 of USNRC Generic Letter 86-10 states "Section III.G.1.b contains the requirements for normal shutdown modes utilizing the control room or emergency control station(s) capabilities. The fire areas falling under the requirements of III.G.1.b are those for which an alternate or dedicated shutdown capability is not being provided." Because III.G.1.b defines alternative by exclusion, GE was requested to define the *original* (normal & redundant) systems. To determine what BWR capabilities constitute the "normal shutdown modes utilizing the control room or emergency control station(s) capabilities," GE was requested to prepare a report covering the *original* (normal and redundant) safe shutdown paths in the GE BWR designs. The report identifies the *original* paths applicable to 10 CFR 50 Appendix R Section III.G.1, relative to one train of systems to achieve and maintain safe shutdown conditions, and include addressing the ADS+LPCI/CS shutdown path. Any of the *original* (normal & redundant) systems can be used as redundant systems to satisfy Appendix R.

Table 1 provides a list of acronyms and abbreviations used in this report.

2. Methods And Assumptions

The evaluations, results and conclusions documented herein are intended to be generic. For completeness, some product line specific equipment and systems are addressed. It is not the intent of the evaluations to perform any design quality analyses. Therefore, the information presented already should exist, and thus, is taken from existing generic documentation.

The equipment and systems information primarily comes from (1) a NRC approved licensing topical report (Reference 2) that generically addresses BWR cooling and depressurization capabilities, and (2) the generic BWR emergency procedure guidelines (Reference 3).

ANS/ANSI-52.1-1983 provides the following generic definition of safe shutdown. A safe shutdown is a shutdown with *(1) the reactivity of the reactor kept to a margin below criticality consistent with technical specifications, (2) the core decay heat being removed at a controlled rate sufficient to prevent core or reactor coolant system thermal design limits from being exceeded, (3) components and systems necessary to maintain these conditions operating within their design limits, and (4) components and systems, necessary to keep doses within prescribed limits operating properly.*

Consistent definitions of cold and hot (safe) shutdown are needed to generically perform the evaluations. The Technical Specification definitions of cold shutdown vary from plant to plant. However, this report is based on compliance to 10 CFR 50, App. R, which was mostly finalized in 1979. Therefore, the definitions of cold and hot shutdown used in this report are in-part based on the NRC's Standard Technical Specifications for the BWR (Reference 4) that were in affect in 1979.

A hot shutdown is defined as a *safe shutdown with the average reactor coolant temperature > 212°F.*

A cold shutdown is defined as a *safe shutdown with the average reactor coolant temperature* ≤ 212°F.

3. Evaluations Of Paths For Achieving Safe Shutdown[1][2]

In addressing the effects of fires on an operating nuclear power plant, it is assumed that the plant is initially operating at 100% power with all systems and components fully capable of performing their intended design functions, and with no limiting conditions of operation in effect. With this set of initial conditions, the postulated Appendix R fire is assumed to occur and be detected. The extent of the fire spread and damage is assumed to be extensive enough to result in a reactor scram, either automatic or manual, and to cause damage to every component in the effected fire area. This is regardless of the size of the fire area, the actual potential for fire ignition and spread within the fire area, and the actual ability of the systems and components to perform in such an environment.

Given this set of initial conditions and bounding assumptions, Appendix R requires that the ability to achieve and maintain safe shutdown conditions be demonstrated. It is generally understood that cold shutdown is the ultimate safe shutdown condition, and that, for each fire area, different means may be necessary and used to achieve cold shutdown. This report identifies the multiple paths available, as a part of the original design of the General Electric Boiling Water Reactor, for achieving safe shutdown. Any of these paths represents an acceptable approach for achieving safe shutdown given the Appendix R fire condition.

Due to the severity of the bounding assumptions required for the Appendix R Safe Shutdown Analysis, Section III.G.1 of Appendix R makes provisions for allowing the unit to remain in a stable hot shutdown condition while repairs are made to systems and components required to support the transition to cold shutdown, the ultimate safe shutdown condition. Therefore, the minimum safe shutdown capability required by Appendix R is that systems and components necessary to support the hot shutdown condition be "free of fire damage," until repairs can be made to systems and components required to achieve cold shutdown.

Although the BWR is designed with the capability to remain in a stable hot shutdown condition using some of the paths described in this report, the discussion provided in this report relative to the various paths available is presented in the context of achieving the ultimate and more desired plant condition of cold shutdown. As such, any of the paths described within this report represent acceptable approaches for achieving safe shutdown for the Appendix R fire condition. This is regardless of whether or not cold shutdown is deferred while systems and component repairs are performed or is achieved directly without such repairs.

In addition, due to the diversity of safe shutdown capability contained in the original systems provided for the GE BWR, it is possible that paths different from those described in this report, can be used to satisfy all of the requirements of Appendix R. On a plant-specific basis, the design capability of the systems described in this report can be used in different combinations to satisfy all of the requirements of Appendix R. These different paths would, in most cases, be a by-product of the specific fire damage in a fire area and of the repair schemes selected by the individual licensees. As such, this report does not generically describe all paths that could be developed for achieving and maintaining safe shutdown for all BWRs.

The following describes some of the safe shutdown paths that are available for achieving and maintaining safe shutdown for the GE BWR.

After a postulated fire is detected, the reactor is scrammed, and the basic (overlapping) steps for achieving and maintaining safe shutdown are:

1. achieving and maintaining safe hot shutdown;
2. reducing reactor pressure vessel (RPV) pressure sufficiently (depressurization) for low pressure cooling to function; and
3. establishing and maintaining shutdown cooling so that the bulk reactor coolant temperature is maintained ≤ 212°F (from the [Reference 4] standard definition of cold shutdown, applicable prior to January 1, 1979).

The above steps are overlapping in that some of the functions to perform one step may be needed to perform the next step. The overall path to achieve and maintain cold shutdown is shown in Figure 1. The available (original) systems and paths to provide core cooling are shown in Figure 2.

During normal operation and anticipated operational occurrences (AOOs), the intent of the BWR design is to maintain hot shutdown conditions if power operation is interrupted, with options available to the operators to proceed to cold shutdown, if needed. For abnormal events, such as an Appendix R fire, the BWR design is intended to provide several ways for automatic logic or manual operator actions to achieve the safest reactor condition, cold shutdown, as soon as practical. At all times, however, the operator has several ways available to take the unit to cold shutdown. To this end, the BWR was originally designed to achieve cold shutdown in a reasonable time (e.g., less than two to three days) using systems that are listed in Tables 2 and 3.

The following sections address each of these overlapping steps and the potentially available equipment for accomplishing the needed safe shutdown functions.

3.1 Achieving and Maintaining Hot Shutdown

As soon as the reactor is scrammed, (Technical Specification) hot shutdown is achieved from the viewpoint of core power generation. Safe hot shutdown can usually be maintained as soon as one of the coolant supply systems (see Table 2) is started to restore and/or reestablish reactor coolant level, and the stored and decay heat energy in the RPV is being released to the isolation condenser (IC), the main condenser or the suppression pool.

To remove stored energy and decay heat, steam can be (normally is) vented to the main condenser via the MSLs and the turbine bypass system. Reactor pressure and saturated zone temperature are controlled by the bypass system as long as sufficient steam flow is available from decay heat.

Whether the RPV is isolated or not, the feedwater system will automatically try to maintain coolant level. Some plants have both steam and electrically driven feedwater pumps. Plant operators can pump water from the condenser hotwell to the RPV using condensate, condensate booster or feedwater pumps (depending on reactor pressure). Venting steam to the main condenser and using these pumps is the usual approach to remove stored energy. As desired (or as the available steaming rate decreases) the operators will slowly depressurize the RPV and maintain coolant level. As such, the condensate, condensate booster and/or feedwater pumps are the normal method for maintaining post-scram coolant level.

If all of the above condensate/feedwater pumps are not available, one or more of the other high pressure systems shown in Table 2 could be used to maintain coolant level. The Reactor Core Isolation Cooling (RCIC) or High Pressure Coolant Injection/High Pressure Core Spray (HPCI/HPCS) system is operated to restore and maintain coolant level. The RCIC and HPCI Systems simultaneously remove energy (steam flow to the turbines) and maintain coolant level. In a high pressure scrammed condition with no available condensate/feedwater pumps, RCIC is the system whose capability most closely matches the required makeup flow.

After the reactor is initially shutdown (scrammed), RCIC or HPCI can maintain coolant level. However, the steamflow to drive the RCIC or HPCI turbine will be less than the steam flow generated by reactor decay heat. Consequently, the SRVs will cycle (open and close) to control reactor pressure.

The BWR2 and early BWR3s have an IC. Via natural circulation an IC removes heat from reactor steam and returns the resulting condensate back to the RPV. About a half hour after a scram, an IC can absorb all the core decay heat, and thus, maintain reactor coolant level. An IC can perform safe shutdown functions as long as make-up water is available to its secondary side.

As a redundant option to using the high and medium pressure (make up) pumps, the operator can depressurize the reactor using the Safety Relief Valves (SRVs) and inject with low pressure systems. Some SRVs can be opened individually, as needed, to control the rate of depressurization, or the operator can manually initiate the Automatic Depressurization System (ADS), which opens a pre-selected group of SRVs which will depressurize the vessel. The SRVs release steam into the suppression pool. SRV actuation will sufficiently reduce pressure to allow the low pressure Core Spray (CS/LPCS) and the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal (RHR) System ECCS cooling loops to restore and maintain reactor coolant level. If the RPV is isolated from the main condenser, then manual initiation of a Residual Heat Removal (RHR) loop in suppression pool cooling mode (with associated auxiliary systems) usually will be required to maintain satisfactory suppression pool temperature. In this configuration, hot shutdown can be maintained for an extended period of time by maintaining reactor pressure below the shutoff head for the low pressure systems but greater than 0 psig.

When sufficient time (estimated ≤ 7 hours) has elapsed, the decay heat boil-off rate in the RPV becomes less than the rated capacity of the control rod drive (CRD) pumps. At this point, the CRD pumps alone can be used to maintain coolant level.

3.2 RPV Depressurization

The uses of the main condenser and ADS with the SRVs to depressurize the RPV are discussed above. However, these are not the only available depressurization paths. The BWR2s and some BWR3s have an isolation condenser (IC) that, with its water make-up supply available, may be used to reduce the RPV pressure. The BWR4s, 5s, 6s and some of the BWR3s have a steam turbine driven RCIC system, which can be aligned to maintain reactor coolant level or provide RPV pressure control through system operation in the test mode. The BWR4s and some of the BWR3s have a steam turbine driven HPCI system, which also can be aligned to maintain reactor coolant level or control RPV pressure through system operation in the test mode. Some RHR systems are capable of operating in steam condensing mode, which can be used to depressurize the RPV from full normal operating pressure to 150 psia.

When an IC is provided, it is piped to the RPV. Via natural circulation, it takes and condenses steam directly from the RPV, and it returns the condensate back to the RPV. The IC typically can absorb the reactor stored energy and decay heat for about 30 minutes, before make-up water to the IC is needed. If IC make-up is available, the IC can be used to reduce the RPV pressure to within the range of the low pressure cooling systems.

RCIC and HPCI systems both use reactor steam to power turbine driven coolant pumps. After the steam loses some energy operating the turbine, it is piped to the suppression pool. The passing of reactor steam via the RCIC/HPCI turbine to the suppression pool results in lowering the RPV pressure. As fission products decay to stable elements, the core decay heat generation rate will naturally decrease over time. When the core decay heat rate has sufficiently decreased (typically one hour after shutdown for HPCI and eight hours after shutdown for RCIC), the steam flow rate that powers the RCIC/HPCI turbine will reach equilibrium with core steam production. (Individual plant calculations are required to determine the equilibrium time points on a plant-specific basis.) From this point on, the RCIC/HPCI system can be used to depressurize the RPV.

The RCIC and HPCI systems normally take coolant from the condensate storage tank (CST), although suppression pool water also may be used, and pump the coolant into the RPV to maintain coolant level. (CST level, suppression pool level and suppression pool temperature are used to determine the water source for the RCIC/HPCI.) Both of these systems can be operated manually. With reactor coolant level being maintained, one or both of these systems can be put into test mode, where the reactor steam is used to pump water from and back to the condensate storage tank. Again, the steam exiting the turbine(s) is piped to the suppression pool. Therefore, assuming the RHR system is maintaining suppression pool temperature, either a RCIC or HPCI system can be used to reduce the RPV pressure to within the range of the low pressure cooling systems. Obviously, the rate of depressurization would be faster with the larger HPCI System.

During RPV depressurization, if containment cooling system(s) (normally used during plant operation) are not available, the RHR system in suppression pool cooling mode will be needed to maintain the suppression pool temperature. Plus, to allow for long-term RCIC/HPCI operation the suppression pool temperature must be maintained below the system's pump temperature limit.

For the few plants that have the RHR steam condensing mode capability, reactor steam can be directed to one or both of the RHR heat exchangers, which remove heat to condense the steam. The resulting condensate can be returned to the reactor via the RCIC system (if condensate is ≤ the RCIC pump temperature limit) or sent to the suppression pool. The RHR system can condense steam with the reactor at its nominal full power operating pressure. About 90 minutes after reactor shutdown, a typical single RHR loop in steam condensing mode can remove enough decay heat (steam) to maintain constant reactor pressure. If two RHR loops are available, it takes approximately 30 minutes after reactor shutdown, before the RHR can remove enough decay heat (steam) to maintain constant reactor pressure. As decay heat continues to decrease, the RHR loop(s), by condensing reactor steam, can reduce reactor pressure (depressurize the RPV) down to 150 psia. However, with time, one RHR loop will have to be switched to suppression pool cooling mode to control pool temperature. Also, one RHR loop (when another water source is not available) may have to be periodically switched to LPCI mode to maintain coolant level, and then returned to steam condensing or suppression pool cooling mode. The priority of RHR modes should be based on first maintaining reactor coolant level sufficient to cool the fuel, second maintaining the suppression pool temperature to be less than its design limit for temperature or net positive suction head (NPSH) requirements for the RHR pumps, and third depressurizing the RPV.

With one or more SRVs passing decay heat steam to the suppression pool, with RHR heat exchanger(s) removing sufficient heat from the suppression pool, and with one or more of the pumps listed in Table 2 maintaining reactor coolant level, the BWR design does allow the BWR to almost indefinitely maintain safe, isolated hot shutdown conditions. If the plant is not intended to be returned to power operation in a reasonable period of time, then plant progression to cold shutdown is the normal near term objective.

3.3 Establishing & Maintaining Cold Shutdown

With the RPV completely depressurized (to atmospheric conditions) and reactor coolant level being maintained, the bulk reactor coolant temperature cannot be greater than 212°F. The BWR has at least four paths for establishing and maintaining bulk reactor coolant temperature ≤ 212°F, and thus, achieve and maintain cold shutdown. Bulk coolant temperature (cold shutdown) can be established and maintained (1) with steam being passed to the main condenser through the bypass valve(s), and any available pump supplying cooling water to maintain coolant level, (2) via one of the two RHR shutdown cooling loops, (3) with the vessel vented to the suppression pool to maintain atmospheric pressure, a low pressure pump maintaining coolant level and bulk reactor coolant ≤ 212°F (which avoids pressurization), and decay heat being removed via a RHR heat exchanger (suppression pool cooling mode), or (4) with one or more SRVs open, a LPCI or CS pump can flood the RPV, coolant would flow back to the suppression pool via the SRV discharge line(s), and decay heat can be removed via a RHR heat exchanger. A general description of each of these paths is discussed below, along with another (long-term only) method of removing decay heat.

1. The normal path for achieving and maintaining cold conditions uses the main condenser. Steam (generated by decay heat) can be vented to the condenser to keep the RPV depressurized. Any one or more of the available pumps (see Table 2) can supply cooling water to maintain coolant level and bulk reactor coolant temperature. If the pump(s) is/are taking water from the main condenser, this path can be used indefinitely. However, if the pump(s) is/are taking water from the suppression pool or condensate storage tank, the source will eventually need to be replenished. Normally, this path would use a RHR loop in the shutdown cooling mode in bringing the unit to cold shutdown.

2. A common path for achieving and maintaining cold conditions is by using the RHR system in shutdown cooling mode. Except for the suction line from only one of the recirculation loops, the typical RHR system has two completely redundant 100% capacity loops. Only if both of these loops are not available, would one of the other paths be used.

3. Another path exists where steam can be vented to the suppression pool by one of the methods described in Section 3.2 to keep the RPV depressurized. A LPCI pump with or without a heat exchanger or a CS pump can maintain coolant level and bulk reactor coolant temperature. One of the two RHR suppression pool cooling loops can be used to remove decay heat.

4. Another path opens one or more SRVs (feasible at atmospheric pressure for all BWR2s, BWR5s and BWR6s, and some BWR3s and BWR4s). A LPCI (with or without a heat exchanger) or CS pump can be used to flood the RPV and MSLs to the level of the open SRV(s). Coolant would then flow back to the suppression pool via the SRV discharge line(s). The decay heat can be removed via a RHR heat exchanger. This path establishes a cooling loop equivalent to a RHR shutdown cooling loop.

As described above, the BWR as originally designed usually has a minimum of four normal and redundant paths to achieve and maintain cold shutdown conditions, and has another redundant path for maintaining cold shutdown conditions on a long-term basis. For example, Table 4 lists some (5+) of the safe shutdown paths applicable to a typical BWR4. Plus, a plant with an IC, if secondary side make-up water is available, can maintain a reactor coolant range from 212°F to slightly above 212°F.

4. Conclusions

The BWR as originally designed has multiple success paths (*original* shutdown capabilities) to achieve and maintain safe (cold) shutdown. The BWR was designed with numerous pumps and various diverse paths for removing stored energy, removing decay heat, and maintaining reactor coolant level, and therefore, achieve and maintain safe shutdown. When any of the normal or redundant paths described in this report are used to achieve and maintain safe shutdown, adequate core cooling is assured.

The operating procedures for normal and abnormal conditions direct the plant operators in the appropriate use of these systems based on observed plant conditions. For example, if high pressure or intermediate pressure coolant make up systems are unavailable, the operator is instructed by the EPG's to manually depressurize the reactor using ADS, and to inject water into the reactor using the available low pressure systems. In this situation, ADS plus a low pressure cooling system (ADS+LPCI/CS) is considered redundant to HPCI/HPCS/RCIC and any of the other paths which rely upon high or intermediate pressure make up systems. For all BWRs, adequate core cooling is maintained, when ADS is used in conjunction with any available low pressure make up system.

5. References

(1) USNRC, "Implementation of Fire Protection Requirements," Generic Letter 86-10, April 24, 1986.

(2) General Electric, "Additional Information Required For NRC Staff Generic Report On Boiling Water Reactors," NEDO-24708A (NRC approved), Revision 1, December 1980.

(3) General Electric and Operations Engineering, "BWR Owners Group Emergency Procedure Guidelines," NEDO-31331, Revision 4, March 1987.

(4) USNRC, "Standard Technical Specifications For General Electric Boiling Water Reactors," NUREG-0123, Revision 2, August 1979.

Table 1

Acronyms & Abbreviations

ADS	Automatic depressurization system
AOO	Anticipated operational occurrence
BWR	Boiling Water Reactor
Cond.	Condenser
CRD	Control Rod Drive (system)
CS	Core Spray (low pressure) (system)
CST	Condensate storage tank
ECCS	Emergency core cooling system(s)
GE	General Electric (Company)
GL	Generic Letter
HPCI	High Pressure Coolant Injection (system)
HPCS	High Pressure Core Spray (system)
IC	Isolation Condenser (system)
LPCI	Low Pressure Coolant Injection (mode of RHR system)
LPCS	Low Pressure Core Spray (system)
MSIV(s)	Main Steamline Isolation Valve(s)
MSL(s)	Main steam line(s)
NRC	Nuclear Regulatory Commission
RCIC	Reactor Core Isolation Cooling (system)
RHR	Residual Heat Removal (system)
RHRSW	Residual Heat Removal Service Water (system)
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup (system)
Rx	Reactor
SDC	Shutdown cooling (mode of RHR system)
SFC	Single Failure Criteria
SPC	Suppression pool cooling (mode of RHR system)
SRV	Safety/Relief Valves
SSC	Structure(s), system(s) and/or component(s)

Table 2

As-Designed Coolant Supply Systems/Components

(Not intended to specify minimum requirements)

System/Component	Pressure Range	Comment(s)
Condensate/Booster	Med to Low*	Plant-specific evaluation required to determine extent of pressure range capability.
Feedwater	High to Low*	Usually must have adequate steam supply, or be electrically powered for post-scram operation. Plant-specific evaluation required to determine extent of pressure range capability.
CRD pumps	High to Low	Usually can maintain RPV coolant level 7 hours (or less) after scram (plant-specific calculation required to determine actual time).
IC condensate (BWR2s & some 3s)	High to Low	Usually can maintain RPV coolant level ½ hour after scram (plant-specific calculation required to determine actual time).
HPCI (Not in BWR5s & 6s)	High to Med	Pressure range overlaps LPCI and (low pressure) CS operating pressure ranges.
RCIC (Not in BWR2s & some 3s)	High to Med	Pressure range overlaps LPCI and (low pressure) CS operating pressure ranges.
HPCS	High to Low	BWR5s & 6s only.
LPCI loops	Low	Pressure range overlaps HPCI and RCIC operating pressure ranges.
low pressure CS loops	Low	Pressure range overlaps HPCI and RCIC operating pressure ranges.

* The condensate and condensate booster pumps are considered as part of the high pressure supply configuration including the main feedwater pumps (only one train of pumps is needed). In the medium pressure range, the FW pumps are not needed, and only a condensate pump and/or a booster pump is/are needed to maintain coolant level in the RPV. In the low pressure range, the FW and booster pumps are not needed, and only a condensate pump is needed to maintain coolant level.

Table 3

As-Designed Systems/Components To Depressurize* RPV
(Not intended to specify minimum requirements)

System/Component	Comment(s)
Main condenser	Via any MSL and turbine bypass.
IC (BWR2s & some 3s)	If make-up water is available to the secondary side of the IC.
RHR	Steam Condensing Mode
ADS or SRV actuation	
RCIC (some BWR3s, all BWR4s, 5s & 6s)	In core cooling or test mode. RHR in suppression pool cooling mode may also be needed. Plant-specific evaluation required.
HPCI (some BWR3s & all BWR4s)	In core cooling or test mode. RHR in suppression pool cooling mode may also be needed. Plant-specific evaluation required.
Reactor Head Vent	Can be used on a long-term basis to maintain atmospheric pressure within the RPV.

* Reduce pressure within the range of the condensate, condensate booster, CS/LPCS and RHR (LPCI and shutdown cooling modes) pumps.

Table 4

Typical BWR4 Safe (Cold) Shutdown Paths

(Other paths also may be available)

Primary Safe Shutdown Paths					
Shutdown Function	**Path 1**	**Path 2**	**Path 3**	**Path 4a/b**	**Path 5a/b**
Reactivity Control	Rx. Scram	Rx. Scram	Rx. Scram	Rx. Scram	Rx. Scram
Pressure Control	Main Cond.	RCIC/SRV	HPCI/SRV	ADS/SRV	ADS/SRV
Inventory Control	Feedwater, Cond., CRD	RCIC	HPCI	LPCI/CS	LPCI/CS
Decay Heat Removal	Main Cond.	RHR SPC	RHR SPC	RHR SPC	RHR SPC
		RHR SDC	RHR SDC	RHR SDC	SRV+RHR SPC*

* SRV+RHR SPC uses either a LPCI or CS Pump to flood the RPV and MSL to the level of the open SRV(s). This allows coolant to flow back to the suppression pool. Decay heat is removed by RHR in the SPC mode.

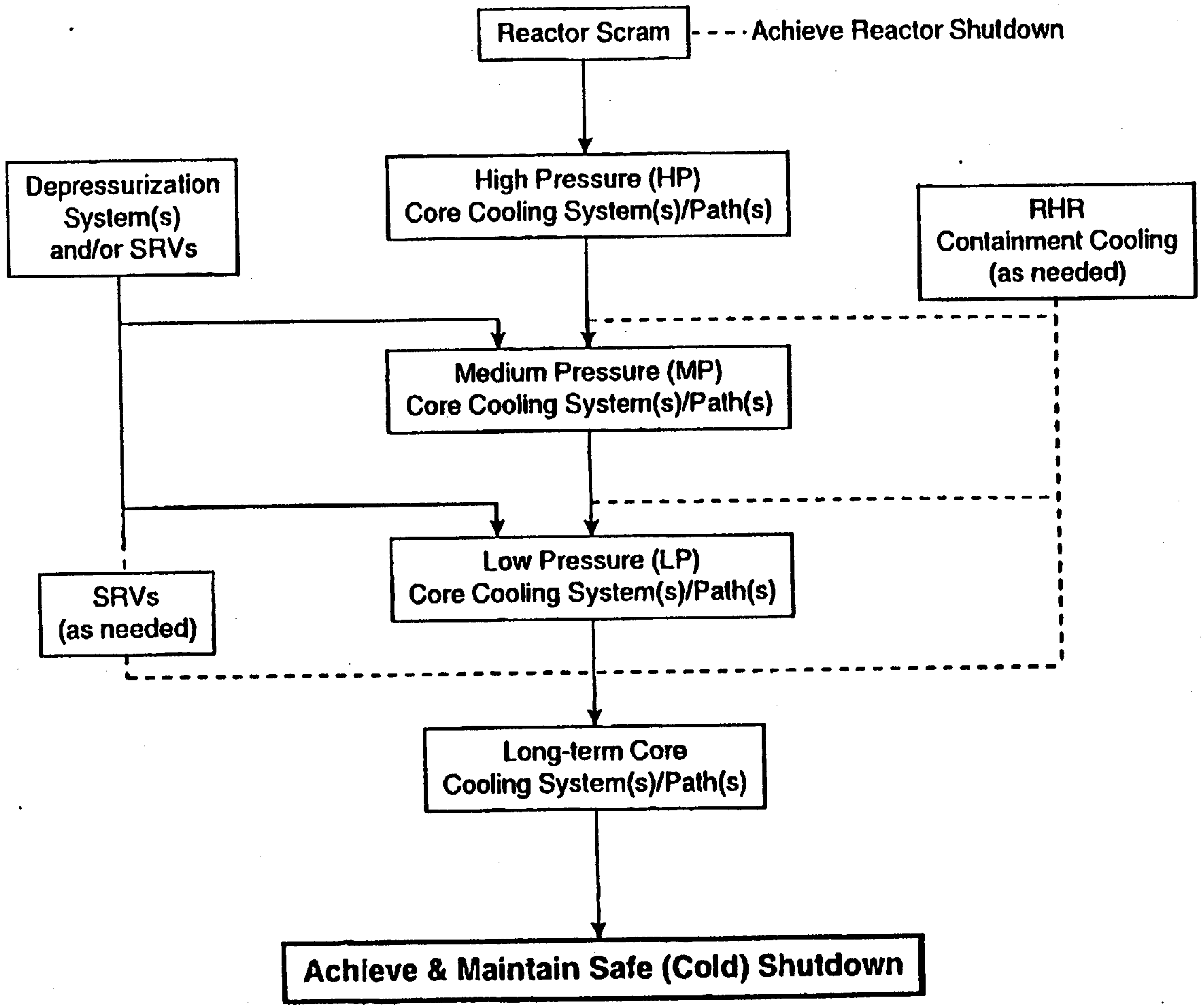


Figure 1

Overall Cold Shutdown Path

Figure 2

Core Cooling Systems & Paths