NRC FORM 464 Part I
(6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA	RESPONSE NUMBER
2000-0202	1

RESPONSE TYPE: [x] FINAL [ ] PARTIAL

REQUESTER: Ms. Nancy Chapman

DATE: APR 26 2000

PART I. -- INFORMATION RELEASED

- [ ] No additional agency records subject to the request have been located.
- [ ] Requested records are available through another public distribution program. See Comments section.
- [ ] APPENDICES _____ Agency records subject to the request that are identified in the listed appendices are already available for public inspection and copying at the NRC Public Document Room.
- [x] APPENDICES A Agency records subject to the request that are identified in the listed appendices are being made available for public inspection and copying at the NRC Public Document Room.
- [ ] Enclosed is information on how you may obtain access to and the charges for copying records located at the NRC Public Document Room, 2120 L Street, NW, Washington, DC.
- [x] APPENDICES A Agency records subject to the request are enclosed.
- [ ] Records subject to the request that contain information originated by or of interest to another Federal agency have been referred to that agency (see comments section) for a disclosure determination and direct response to you.
- [ ] We are continuing to process your request.
- [ ] See Comments.

PART I.A -- FEES

AMOUNT *: $ 54.55

* See comments for details

- [x] You will be billed by NRC for the amount listed.
- [ ] You will receive a refund for the amount listed.
- [ ] None. Minimum fee threshold not met.
- [ ] Fees waived.

PART I.B -- INFORMATION NOT LOCATED OR WITHHELD FROM DISCLOSURE

- [ ] No agency records subject to the request have been located.
- [ ] Certain information in the requested records is being withheld from disclosure pursuant to the exemptions described in and for the reasons stated in Part II.
- [ ] This determination may be appealed within 30 days by writing to the FOIA/PA Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Clearly state on the envelope and in the letter that it is a "FOIA/PA Appeal."

PART I.C COMMENTS (Use attached Comments continuation page if required)

The actual fees for the processing of your request are:

Professional Search	**- 1 hr. @ $39.00 per hr.**	**=**	**$39.00**
Professional Review	**- 15 mins. @ $0.65 per min.**	**=**	**9.75**
Duplication	**- 29 pgs. @ $0.20 per pg.**	**=**	**5.80**
Total			**$54.55**

In a telephone conversation with Nina Pugh on April 25, 2000, you agreed to pay the fees noted above. This completes NRC's action on your request.

SIGNATURE - FREEDOM OF INFORMATION ACT AND PRIVACY ACT OFFICER

Carol Ann Reed *Carol Ann Reed*

FOIA No. 2000-0202

APPENDIX A
RECORDS BEING RELEASED IN THEIR ENTIRETY
(If copyrighted identify with *)

NO.	DATE	DESCRIPTION/(PAGE COUNT)
1.	06/23/86	Memo to T. Novak, NRC from W. Russell, NRC Re: WOG Emergency Response Guidelines Supplemental Safety Evaluation of Revision 1 (29 pages)

JUN 2 3 1986

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MEMORANDUM FOR: Thomas M. Novak, Acting Director
Division of PWR Licensing-A

FROM: William T. Russell, Director
Division of Human Factors Technology

SUBJECT: WOG EMERGENCY RESPONSE GUIDELINES SUPPLEMENTAL SAFETY EVALUATION OF REVISION 1

We have reviewed Revision 1 of the Westinghouse Owners' Group Emergency Response Guidelines (ERGs). Our Supplemental Safety Evaluation Report is enclosed.

The staff had previously reviewed the BASIC version of the ERGs and reviewed Revision 1 of the ERGs for changes from the BASIC version that were made in response to the staff's comments on the BASIC version. The enclosed report covers our review of the additional differences between the BASIC version and Revision 1. In discussing the generic issues and individual guidelines, this report provides amplification of some of the issues in our previous SER and points out that with regard to Pressurized Thermal Shock and Stagnant Reactor Coolant Loops, the staff has not completed its review and therefore changes may be necessary to the related guidelines FR-P.1 and FR-P.2.

Based on this review, we have concluded that Revision 1 of the ERGs is a substantial improvement over the BASIC version and that implementation of Revision 1 should continue while the staff's outstanding comments are resolved as part of the program for maintenance of the ERGs.

We request that you transmit the enclosed letter, with the SSER, to the WOG.

This SSER was prepared by S. D. MacKay, Senior Nuclear Engineer. Any questions regarding this SSER should be referred to Mr. MacKay (x24875).

William T. Russell, Director
Division of Human Factors Technology

Enclosures:
As stated

SDM/MEMO FOR HLTHOMPSON

A/1

OFFICE	HFIB	HFIB:ABC	DHFT:ADD	DHFT:D			
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Mr. D. Butterfield, Chairman
Westinghouse Owners' Group
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Butterfield:

SUBJECT: WOG EMERGENCY RESPONSE GUIDELINES SUPPLEMENTAL SAFETY EVALUATION OF REVISION 1

We have reviewed Revision 1 of the Westinghouse Owners' Group Emergency Response Guidelines (ERGs). Our Supplemental Safety Evaluation Report is enclosed.

The staff had previously reviewed the BASIC version of the ERGs and reviewed Revision 1 of the ERGs for changes from the BASIC version that were made in response to the staff's comments on the BASIC version. The enclosed report covers our review of the additional differences between the BASIC version and Revision 1. In discussing the generic issues and individual guidelines, this report provides amplification of some of the issues in our previous SER and points out that with regard to Pressurized Thermal Shock and Stagnant Reactor Coolant Loops, the staff has not completed its review and therefore changes may be necessary to the related guidelines FR-P.1 and FR-P.2.

Based on this review, we have concluded that Revision 1 of the ERGs is a substantial improvement over the BASIC version and that implementation of Revision 1 should continue while the staff's outstanding comments are resolved as part of the program for maintenance of the ERGs.

Sincerely,

Original signed by:
Thomas M. Novak

Thomas M. Novak, Acting Director
Division of PWR Licensing-A
Office of Nuclear Reactor Regulation

Enclosure:
Supplemental Safety Evaluation Report on Westinghouse Owners' Group Emergency Response Guidelines

SDM/MEMO FOR HLTHOMPSON

OFFICE	HFIB	HFIB:ABC	DHFT:ADD	DHFT:D	PWR-A:ADD	PWR-A:AD
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DATE	6/23/86	6/24/86	6/26/86	6/26/86	6/ /86	6/7/86

SUPPLEMENTAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

IN THE MATTER OF

REVISION 1 OF THE

WESTINGHOUSE OWNERS GROUP

EMERGENCY RESPONSE GUIDELINES

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SUPPLEMENTAL SAFETY EVALUATION

OF

WESTINGHOUSE OWNERS GROUP

EMERGENCY RESPONSE GUIDELINES

1.0 INTRODUCTION

The purpose of this report is to discuss the differences between the BASIC version and Revision 1 of the Westinghouse Emergency Response Guidelines (ERGs). Revision 1 contains several significant improvements over the BASIC version including the addition of background material on several generic issues, consolidation of the operator's response to a loss of reactor coolant and to a loss of secondary coolant into a single guideline, improved safety injection termination criteria, additional steam generator tube rupture analyses, and a determination of the applicability of the guidelines for accidents occurring under conditions other than full power. Revision 1 also contains a configuration control and approval procedure that is being developed to accommodate future comments and implement changes to the ERGs.

2.0 BACKGROUND

On June 3, 1983 the NRC Division of Licensing issued Generic Letter No. 83-22, which promulgated the staff's June 1, 1983 Safety Evaluation of the BASIC version of the Westinghouse Owners Group (WOG) Emergency Response Guidelines (ERGs). The Safety Evaluation presented the staff findings and listed 45 items to be resolved by the issuance of Revised Guidelines or by a Supplemental Safety Evaluation.

By letter dated September 27, 1983, (Reference 2) the Westinghouse Owners Group responded to our Safety Evaluation Report (SER). With regard to the four unresolved items related to NUREG-0737, the WOG stated that Revision 1 would address the two items designated by the staff for resolution in Revision 1. (Additional information was subsequently provided for the other two items and evaluated by the staff in Reference 7.) The letter provided a response to each of the 28 generic items in our SER and stated that the 12 plant-specific items would be addressed in Revision 1 where appropriate.

The letter also included an enclosure; "Configuration Control and Approval Procedure for Guidelines and Background Documents."

By letter dated November 30, 1983, (Reference 3) the WOG submitted Revision 1 of the Emergency Response Guidelines. These guidelines included: Optimal Recovery Guidelines, Fold Out Criteria, Contingency Action Guidelines, Critical Safety Function Status Trees, and Function Restoration Guidelines.

By letter dated May 4, 1984 (Reference 4), the WOG submitted the bases for these Guidelines. The bases consist of an Executive Summary (one volume), five volumes of background material for plants with high pressure safety injection and five volumes of background material for plants with low pressure safety injection. High pressure safety injection systems are capable of pumping water into the Reactor Coolant System at pressures greater than the high pressure set point of the pressurizer relief valves whereas the "low pressure plants" are capable of Safety Injection at maximum pressures of approximately 1500 psig.

By letter dated August 15, 1984, (Reference 5) the WOG provided a comparison of Revision 1 of the ERGs with the earlier BASIC version. Based on our review of this comparison, the staff gave preliminary approval of Revision 1 for implementation. This approval was issued on December 27, 1984, (Reference 6) with the understanding that there remained three categories of items to be resolved on a longer-term schedule. They were: (1) outstanding items from our Safety Evaluation (Reference 1) of the BASIC version, (2) recommendations for enhancement from the ERG Validation Program (Reference 11), and (3) comments on Revision 1 of the ERGs resulting from our longer term detailed review.

The NRC staff then reviewed those portions of the Revision 1 guidelines where changes from the BASIC version had been made in response to the staff's safety evaluation of June 1, 1983. Based on this review, the

staff issued its Safety Evaluation of December 26, 1985 (Reference 7). In that Safety Evaluation, the staff concluded that 34 of the 45 open items had been resolved by the issuance of Revision 1 of the Emergency Response Guidelines and that the staff would seek further improvements in eleven areas carried over from its review of the BASIC version of the ERGs. On February 20, 1986, the NRC staff met with the WOG and discussed the relevant areas of concern. The WOG will provide a formal response to the staff's concerns.

The remainder of the staff's review of Revision 1 of the ERGs involved the differences between the BASIC version and Revision 1 that were not the result of the staff's comments but were based on other considerations including nuclear plant simulator testing of the BASIC version of the ERGs, a review of Pressurized Thermal Shock, comments from Electric Utilities, a review of pertinent INPO documents and evaluation of the Ginna Steam Generator Tube Rupture Event. These differences between the BASIC version and Revision 1 of the ERGs are discussed below.

3.0 EVALUATION

3.1 Administrative Changes

3.1.1 Writer's Guide and User's Guide

Revision 1 contains a detailed Writer's Guide that explains the two-column format of the ERGs and establishes the methods of presenting the Optimal Recovery Guidelines (ORGs), Functional Recovery Guidelines (FRGs) and Critical Safety Function Status Trees. It also contains a User's Guide that establishes priorities among the ORGs and FRGs and how they are used in conjunction with the Critical Safety Function Status Trees. We have reviewed these guides and found their technical content acceptable.

Our review of these guides was limited to their technical adequacy. Although some human factors techniques were used in the presentation of the ERGs, additional human factors input is necessary to produce acceptable procedures from the ERGs. Since the NRC reviews plant-specific procedures generation packages that describe how licensees will address human factors concerns, it is not necessary for the NRC to review the human factors aspects of the ERGs. Therefore, the ERGs are considered technical guidelines and we have reviewed them on that basis.

3.1.2 Configuration Control

Realizing that the upgrading of the ERGs is a continuing process, the WOG is developing a configuration control and approval procedure to accommodate future changes to the ERGs. It will include a method of establishing the safety significance, priority, and methods of implementing proposed changes. We reviewed the Configuration Control System presented in Revision 1 and found that it needed further development. This matter is discussed in our Safety Evaluation of December 26, 1985 (Reference 7) and will be the subject of a Supplemental Safety Evaluation.

3.1.3 Generic Issues

A significant change from BASIC to Revision 1 is the development and presentation of Generic Issues as separate topics of discussion in the Executive Volume. These topics include: Reactor Coolant Pump Trip and Restart, Safety Injection Termination and Reinitiation, Safety Injection Reduction, Operator Action Criteria on Fold Out Pages, Steam Generator Tube Integrity, Pressurized Thermal Shock, Stagnant Coolant Loops, Natural Circulation, Reactor Coolant System Voiding, Reactor Vessel Liquid Inventory System, Generic Instrumentation, Loop

Stop Valves, Technical Specifications and Quality Assurance Standards of Regulatory Guide 1.33, and ANSI/ANS-3.2. Most of these topics were addressed within individual guidelines in the BASIC version but we have found that the presentation in Revision 1 allows a more comprehensive understanding of the issues and provides a better basis for training on these issues. We therefore find this change acceptable.

3.2 Generic Issues

We have reviewed the Revision 1 background material dealing with generic issues. The most significant changes are discussed below.

3.2.1 Reactor Coolant Pump Trip Criteria

Revision 1 background contains an evaluation of alternate Reactor Coolant Pump (RCP) trip parameters to establish a variable which will reduce the probability of unnecessary RCP trip for SGTRs and non-LOCAs, while still providing for timely RCP trip for small break LOCAs. The results of this evaluation can be used by utilities to establish the appropriate RCP trip parameter and setpoints for use in Plant-Specific Emergency Operating Procedures based on the Emergency Response Guidelines.

The relevant parameters evaluated are Reactor Coolant (RC) Pressure, RC Subcooling and RC-to-Steam Generator (SG) Pressure Differential. The resulting trip parameter selection is expected to result in improved operation of the RC pumps over a wide range of accident conditions.

In NRC Generic Letters 83-10c and 10d, the NRC addressed the questions of developing RCP trip setpoints which do not cause RCP trip for those transients and accidents where forced

circulation and pressurizer pressure control is a major aid to the operator, yet alert the operators to trip the RCPs for those small LOCAs where continued operation and subsequent trip might result in core damage. The NRC concluded that the need for RCP trip following a transient or accident should be determined by each plant considering Owners Group input, and provided guidance for the development of satisfactory RCP trip setpoints. Further guidance was provided by the NRC in Generic Letter 85-12 (Reference 8).

Based on our review, we have concluded that the Revision 1 ERGs together with NRC Generic Letter 85-12 provide adequate guidance for the establishment of Reactor Coolant Pump trip setpoints.

3.2.2 Safety Injection Reduction

In response to transients which require the reduction of Safety Injection (SI) flow in combination with cooldown and depressurization of the RCS, an SI Reduction Sequence is accomplished in Revision 1 by a series of pump trips. The BASIC version of the guidelines required the "throttling" of SI flow. This change to the guidelines is desirable because the throttling would be typically performed by an operator stationed at the valve location which could become a high radiation area under accident conditions. Tripping pumps will accomplish the desired flow reduction and therefore is the preferred method.

Revision 1 presents the methodology for calculating appropriate subcooling criteria, and for evaluating on a plant-specific basis, a sequence for reducing safety injection flow in coincidence with RCS cooldown. The REDUCE computer model was developed by Westinghouse for the utilities to use in the determination of the amount of subcooling required prior to the

reduction of SI flow by shutting off an SI pump. The REDUCE code was developed to determine the optimum use of SI pumps during cooldown and depressurization for such events as the Steam Generator Tube Rupture with Loss of Reactor (or Secondary) Coolant. The optimization was to provide minimal cycling of the SI pumps and minimum loss of water from the Reactor Coolant System. However, the resultant procedures involved three different sets of ECCS reduction criteria with a total of 42 individual subcooling margin requirements. The staff is concerned that the procedures may be confusing to the operator or occupy more of his/her time than is justified by the benefits gained.

The WOG has been requested to review the benefits of these procedures and consider simplifying the criteria and reducing the number of subcooling requirements. This is an open item in our Supplemental Safety Evaluation of December 26, 1985 (Reference 7).

3.2.3 Safety Injection Termination Criteria

The SI Termination Criteria in BASIC included, as a necessary condition for SI termination, that Reactor Coolant (RC) pressure achieve either an established value or a 200 psi increase. In Revision 1, this criterion was reduced to "stable or increasing" pressure to avoid unnecessary flow of RC into the steam generator. Since the modified criteria still require adequate RC subcooling, RC inventory and Secondary heat sink, the modified criteria provide adequate assurance of core cooling under accident and transient conditions, and are therefore acceptable.

3.2.4 Pressurized Thermal Shock and Stagnant Reactor Coolant Loops

The background material in Revision 1 is supplemented by considerations from WCAP-10319. "A Generic Assessment of Significant Flaw Extension, Including Stagnant Loop Conditions, From Pressurized Thermal Shock of Reactor Vessels in Westinghouse Nuclear Power Plants" (Reference 10). The guidelines for operator response have also been changed and these are discussed later in this report. The staff is continuing its review of these two related generic issues and the results of this review will be reported in a subsequent Safety Evaluation Report.

3.3 Emergency Response Guidelines

The Emergency Response Guidelines in Revision 1 consist of 24 Optimal Recovery Guidelines, 6 Critical Safety Function Status Trees, and 18 Function Restoration Guidelines. Several of the changes to these guidelines were reviewed previously and the results were reported in our Supplemental Safety Evaluation Report of December 26, 1985 (Reference 7). Some of the issues remain unresolved. All of the guidelines are accounted for below.

3.3.1 Optimal Recovery Guideline Changes

E-0, Reactor Trip or Safety Injection

This guideline remains the "immediate action and diagnostic" guideline, just as in BASIC. Minor changes in the transitions have been made to be consistent with the overall renumbering of the individual guidelines. A new transition has been added for the Revision 1 guidelines ECA-1.2, LOCA OUTSIDE CONTAINMENT, which was not part of the BASIC set. We have reviewed these changes and found them acceptable.

ES-0.0, Rediagnosis

This is a new guideline in Revision 1. It was developed in response to a finding from the BASIC Validation Program. However, the overall consistency of Revision 1 has significantly reduced its usefulness. No transitions exit anywhere in the ERG set to this guideline. It is to be used strictly at the operator's discretion, and should only confirm that he/she is already performing the correct procedure.

We have reviewed this guideline and we have concluded that since the operator is not required to perform this procedure, it will not interfere with his/her performance of the correct actions. Furthermore, this guideline may be of assistance in correctly diagnosing a steam generator tube rupture (SGTR) or a steam line rupture. We therefore find this guideline acceptable.

ES-0.1, Reactor Trip Response

(No SI)

ES-0.2, Natural Circulation Cooldown

(BASIC ES-0.2A)

ES-0.3, Natural Circulation Cooldown With Steam Void in Vessel (W/RVLIS)

(BASIC ES-0.2B)

ES-0.4, Natural Circulation Cooldown With Steam Void in Vessel (W/O RVLIS)

(BASIC ES-0.2C)

Guidelines ES-0.1, ES-0.2, ES-0.3, and ES-0.4 are essentially unchanged from the BASIC version.

E-1, Loss of Reactor or Secondary Coolant

This guideline is a consolidation of the BASIC guidelines E-1: Loss of Reactor Coolant and E-2: Loss of Secondary Coolant. This change was a natural consequence of the development of a single SI Termination criteria set and the similarity of actions for a LOCA or a secondary break.

We reviewed this guideline with consideration of the symptoms available to the operator in the control room from a Loss of Reactor Coolant and a Loss of Secondary Coolant and the corrective actions required for each event. We found that the symptoms and required actions are the same for the initial stages of either event and that appropriate diagnostic steps direct the operator to other guidelines when more event-specific action would be required such as main steam line isolation or transfer to cold leg recirculation. Based on these considerations, we found this guideline acceptable.

ES-1.1, SI Termination

In the BASIC version, three separate guidelines were provided for terminating SI: ES-0.3, ES-1.1, and ES-2.1. Each of these guidelines did basically the same thing; however, the check for SI reinitiation was different because of the different termination criteria. With the standardization of SI termination/reinitiation criteria in Revision 1, the need for multiple guidelines for termination was eliminated. This is the only SI Termination Guideline in Revision 1.

We have reviewed the revised SI Termination criteria and found them acceptable as discussed above in Section 3.2.3. We also found that having only one guideline for SI Termination enhances the useability of the ERGs. Based on these considerations we found this guideline to be acceptable.

ES-1.2, Post-LOCA Cooldown and Depressurization

This guideline corresponds to the ES-1.2 guideline in the BASIC set. The replacement of the "throttled" type of SI reduction with the "stepwise" technique, which is presented in detail in a Generic Issues document, was a major contributor to the Revision 1 guideline. New analyses showing expected plant response to the actions of this guideline are included in the background document.

The staff is concerned that the large number of subcooling margin requirements may be confusing to the operator or occupy more of his/her time than is justified by the benefits gained. This is discussed above in Section 3.2.2. The staff is also seeking more consistent criteria and parameter values for accumulator isolation, as explained in Reference 7. These two matters are open items in our Safety Evaluation of December 26, 1985 (Reference 7).

ES-1.3, Transfer to Cold Leg Recirculation

This guideline is equivalent to the BASIC ES-1.3, but also serves to replace the BASIC ES-2.1, which provided the same guidance. Some detail was reduced from the BASIC version because of the very plant-specific nature of the transfer sequence. This allows the detail to be added in the plant-specific procedure without having to take exception to the generic guideline. We found the guideline to be acceptable.

ES-1.4, Transfer to Hot Leg Recirculation

This guideline is equivalent to the BASIC ES-1.4, with some reduction in detail. We consider it appropriate that individual licensees provide the details for establishing flow paths and we find this guideline acceptable.

E-2, Faulted Steam Generator Isolation

This guideline is significantly reduced in size from the BASIC version of E-2: Loss of Secondary Coolant, and now serves only to provide the guidance suggested in its title. The balance of the actions in the BASIC version have been moved to the E-1 guideline in Revision 1 as discussed above. We find this revised E-2 acceptable.

E-3, Steam Generator Tube Rupture

The entire series of SGTR-related guidelines has been restructured to present the recovery philosophy in a more concise manner. The guidance presented in Revision 1 E-3 was previously spread between E-3, ES-3.2, and ECA-3, in the BASIC guidelines. Several steps contain wording changes to clarify the intent. The SI Termination criteria are very similar to the criteria used elsewhere throughout the ERGs. The E-3 guideline has been considerably streamlined by consolidating contingency actions in the ECA-3 series of guidelines, and by completely separating the alternate post-SGTR cooldown guidelines.

The BASIC version required a 200 psi pressure increase in the RCS prior to SI termination in order to assure adequate RCS pressure and inventory control. This requirement is not present in the Revision 1 version.

Additional analyses for offsite power not available and for the effects of coincident cooldown and depressurization on the recovery are included in the background document along with earlier analyses to demonstrate the recovery strategy and to facilitate operator training.

The staff has reviewed this guideline and concluded that the changes made from the BASIC version have improved the guidance for the operator and are acceptable.

ES-3.1, Post-SGTR Cooldown Using Backfill
(BASIC ES-3.2.A)

ES-3.2, Post-SGTR Cooldown Using Blowdown
(BASIC ES-3.2B)

ES-3.3, Post-SGTR Cooldown Using Steam Dump
(Included in BASIC E-3)

ECA-0.0, Loss of All ac Power
(BASIC ECA-2)

ECA-0.1, Loss of All ac Power Recovery Without SI Required
(BASIC ECA-2.1)

ECA-0.2, Loss of All ac Power Recovery With SI Required
(BASIC ECA-2.2)

ECA-1.1, Loss of Emergency Coolant Recirculation
(BASIC ECA-5)

These seven guidelines are essentially unchanged from the BASIC version. However, the matter of accumulator isolation criteria, employed in ECA-0.0 and ECA-1.1, should be resolved as discussed in our Safety Evaluation of December 26, 1985 (Reference 7).

ECA-1.2, LOCA Outside Containment
The BASIC version of the ERGs did not contain an Optimal Recovery Guideline (ORG) for a LOCA outside containment. This

ORG is now part of Revision 1 of the ERGs. It directs the operator to try to isolate the leak and if this fails, the operator is directed to ECA 1.1, "Loss of Emergency Coolant Recirculation." The staff raised two concerns with these guidelines: (1) if the LOCA cannot be contained, the maximum safe cooldown rate and depressurization should be employed in order to minimize releases to the atmosphere, and (2) some of the steps in ECA 1.1 are not applicable to the LOCA outside containment and therefore this should be made clear in the guidelines or a separate guideline should be written.

The WOG will reconsider the present cooldown rate limit of 100°F per hour and explain the balancing of the threat to reactor vessel integrity at higher cooldown rates against the prospect of higher releases with the slower cooldown. The WOG will also review ECA 1.1 for clarification of which steps apply to the LOCA outside containment.

ECA-2.1, Uncontrolled Depressurization of All Steam Generators

This guideline contains the contents of BASIC guidelines ECA-4 and ES-2.3. Special attention is given to the expected occurrence of excessive RCS cooldown and the resultant pressurized thermal shock concern.

Two items remain unresolved in this guideline: the final review of pressurized thermal shock as discussed in Section 3.2.4 above may affect this guideline and the matter of accumulator isolation criteria should be resolved as discussed in our Safety Evaluation of December 26, 1985 (Reference 7).

ECA-3.1, SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired

ECA-3.2, SGTR With Loss of Reactor Coolant-Saturated Recovery Desired

Because of the restructuring of the entire E-3 series of guidelines, this particular guideline contains information from the BASIC guidelines ES-3, ECA-3, ECA-7, and ECA-8.

These guidelines employ the SI reduction sequence for which the staff is seeking simplification as discussed above in Section 3.2.2. Furthermore, difficulties had been encountered with cooldown techniques during a simulator exercise. There is a large procedural loop to be performed in ECA 3.1, "Steam Generator Tube Rupture with Loss of Coolant" in which the operator must perform Steps 18 through 37, then go back and perform Steps 9 through 18, before he is allowed to shut off a second SI pump. The operator failed to shut off the second SI pump in a timely manner, resulting in repressurization and possible steam generator overfill. Performance of the procedural loop, even if recognized by the operator, is not likely to be done in 2.2 minutes as shown in the analysis presented in the background document. There is also a similar loop in ECA-3.2.

These guidelines also contain the questionable accumulator isolation criteria used throughout the ERGs. The SI Reduction Sequence and the accumulator isolation criteria are open items in our Safety Evaluation of December 26, 1985 (Reference 7).

Although not indicated by the title, these guidelines are used in the event of a SGTR with a loss of reactor coolant or with a loss of secondary coolant. The analyses in the background

material for these guidelines are for a SGTR with a loss of secondary coolant. The staff suggests that the titles of these guidelines be changed to indicate more accurately their intended use.

ECA-3.3, SGTR Without Pressurizer Pressure Control

The Revision 1 guideline was designated ECA-9 in the BASIC set and is basically unchanged.

3.3.2 Critical Safety Function Status Trees

Several changes were made in developing the Revision 1 Critical Safety Function Status Trees. Fundamentally, the intent and usage of the status trees remains unchanged from BASIC. However, one change was made in the sequence of Critical Safety Function priority, and thus status tree monitoring. In Revision 1, the Heat Sink tree is third in priority and Integrity is fourth, which is the reverse of the BASIC version. This change was made to reflect the severity of required actions in the RED path Heat Sink response guideline (Bleed and Feed cooling in the RCS) compared to the RED path actions in the Integrity guidelines.

The background material in Revision 1 indicates that for plants with low capacity pressurizer relief valves, the loss of heat sink could lead to a condition of Inadequate Core Cooling (ICC). Because of this close relationship between the loss of heat sink and ICC, the importance of the Heat Sink Critical Safety Function was increased by interchanging the Integrity and Heat Sink functions.

The staff also considered the effect that this interchange might have on the Integrity Critical Safety Function. The relative risk from thermal stresses resulting from various transients is

given in the background material regarding stagnant RC loops. The data presented shows a much lower risk of flaw extension resulting from a loss of heat sink than that which would result from a LOCA or SGTR. This indicates that the actions taken to restore the steam generator heat sink or implement bleed and feed operation should not significantly affect the parameters that signal the need for integrity function actions. Therefore, we find this change in priorities acceptable.

In addition, the Revision 1 Status Trees are presented in both the "block" and "branch" versions, to provide utilities with the additional format option. All setpoint values in the status trees use only ADVERSE CONTAINMENT values for conservative instrument uncertainties; all trees also now present a RED path at the top and GREEN paths at the bottom. We find these changes acceptable.

The question of the completeness of the ERGs and whether a status tree should be added for radiation control is an open item in our Safety Evaluation of December 26, 1985 (Reference 7).

F-0.1, Subcriticality

This tree is modified from the BASIC version to show that the normal, post-trip flux decay rate is a satisfied (GREEN) condition. This merely completes the status tree and is acceptable.

F-0.2, Core Cooling

This tree is essentially the same as BASIC with the consideration of the consolidation of four BASIC guidelines into three Revision 1 guidelines. The adequacy of the reactor vessel

water level action point for inadequate core cooling is questioned in our Safety Evaluation of December 26, 1985 (Reference 7).

F-0.3, Heat Sink

This tree contains a modified RED path symptom, using an indicated minimum feedwater flow or SG level rather than the ambiguous words "feedwater available." It no longer contains ORANGE termini as in the BASIC version because of the reduced emphasis of the associated challenges. The last two termini in the Revision 1 version are also reversed compared to BASIC. This reflects a reassessment of the relative severity of the two conditions. We have reviewed these changes and concluded that they are improvements over the BASIC version.

F-0.4, Integrity

This tree has been modified from the BASIC version to include a path considering Cold Overpressure concerns. Also the plot showing the temperature-pressure domains for PTS has had the color-coding and the normal Technical Specification cooldown curve removed.

Inclusion of the Cold Overpressure concerns is an improvement over the BASIC version and its presentation in this status tree is acceptable. Removal of the color coding from the temperature-pressure plot does not detract from its usefulness. Also, the technical specification limit is not appropriate since operator action is taken after the technical specification limit has been exceeded and requires a soak period or a slower cooldown than would be allowed by the technical specifications. The staff concluded that the Revision 1 Integrity Status Tree is acceptable.

F-0.5, Containment

This tree is unchanged from the BASIC version.

F-0.6, Inventory

This tree is unchanged from the BASIC version.

3.3.3 Function Restoration Guidelines

Because the Function Restoration Guidelines were developed in the latter stages of the BASIC Guidelines Program, they already contained many of the features inherent in Revision 1. The relationship of individual guidelines to their BASIC counterparts and any significant internal changes are discussed below:

FR-S.1, Response to Nuclear Power Generation/ATWS

This guideline in Revision 1 combines the FR-S.1 and ECA-1 (ATWS) guidelines from BASIC. The guidelines title is expanded to show the present dual function.

The background document for this dual-purpose guideline not only reflects the dual functions, but also contains an expanded discussion of the available ATWS analysis. We find these changes acceptable.

FR-S.2, Response to Loss of Core Shutdown

This guideline in Revision 1 is the same as its BASIC counterpart, with the additional considerations due to modifications made to the status tree and is acceptable.

FR-C.1, Response to Inadequate Core Cooling

This guideline is effectively the same as its BASIC counterpart. Operator action criteria have been made more definite and some confusing loops have been eliminated. However, the staff has questioned the adequacy of the reactor vessel water levels for initiating the actions of ERGs FR-C.1 and FR-C.2. The staff is

concerned that the ERG background material may lead operating crews to believe that the 3½ foot water level reading is a safe level under all conditions whereas, the staff's preliminary analysis indicates that core damage could occur at the 3½ foot water level reading when decay heat levels are low, with the consequent froth levels being low due to the lower steam bubble content of the reactor coolant. The ERG background material does not provide a clear basis for the 3½ foot level but implies that the froth level will be high enough to cool the core. The WOG will review the adequacy of the levels for initiating actions under C.1 and C.2, and will consider clarifying the background material.

This guideline also contains the questionable accumulator isolation criteria (Reference 7). Both of these items will be addressed in a supplement to this SER.

FR-C.2, Response to Degraded Core Cooling

This guideline is a consolidation of the FR-C.2 and FR-C.3 guidelines from BASIC. Consistent with the status tree changes, this guideline now is the appropriate response to all ORANGE priority challenges to the safety function. Entry can now be made with or without RCPs operating.

The comments on Guideline FR-C.1 above regarding reactor vessel water level and accumulator isolation also apply to Guideline FR-C.2.

FR-C.3, Response to Saturated Core Cooling

This guideline corresponds to the BASIC FR-C.4 guideline, and is fundamentally unchanged.

FR-H.1, Response to Loss of Secondary Heat Sink

This guideline is unchanged in strategy from its BASIC counterpart except for the inclusion of a SI flow reduction and PORV closure sequence to restore the plant to semi-normal conditions after bleed and feed has been initiated, and after secondary heat sink has been restored. Much greater attention is paid in the background document to the proper symptoms requiring entry into bleed and feed cooling. Also, all RCPs are tripped if auxiliary feedwater cannot be established early in the guideline. RCPs are no longer restarted in this guideline.

New analyses concerning RCP usage, RCS response immediately following loss of all feedwater, and attempts to reach RHR conditions before SG dryout, are contained in the background document.

This guideline may require modification based on the resolution of the SI Reduction issue discussed in Section 3.2.2 above.

FR-H.2, Response to Steam Generator Overpressure

This guideline is unchanged from BASIC.

FR-H.3, Response to Steam Generator High Level

This guideline is unchanged from BASIC.

FR-H.4. Response to Loss of Normal Steam Release Capabilities

This guideline corresponds to FR-H.5 in BASIC and is essentially unchanged from that version. The name was changed from the longer form "Response to Loss of Steam Generator PORVs and Condenser Dump Valves."

FR-H.5, Response to Steam Generator Low Level

This guideline corresponds to FR-H.4 in BASIC. The reversal of H.4 and H.5 in Revision 1 is consistent with the Heat Sink Status Tree modifications and is acceptable.

FR-P.1, Response to Imminent Pressurized Thermal Shock Condition

FR-P.2, Response to Anticipated Pressurized Thermal Shock Condition

The latest BASIC version of these two guidelines was not considered in our Safety Evaluation of June 3, 1983 (Reference 1). The latest BASIC version was subsequently reviewed and is discussed in our Safety Evaluation Report of March 11, 1985 (Reference 9). Changes in these guidelines involve the criteria for requiring a "soaking" period in FR-P.1, the deletion of a post-recovery "soak" in FR-P.2 and the deletion of actions based on limits in the Technical Specifications.

The Revision 1 version of these guidelines is the same as the latest BASIC version except that additions have been made to FR-P.1 to provide for multiple steam generator failures and for a failed open pressurizer relief valve. The staff has not completed its review of the pressurized thermal shock issue and these guidelines may require modifications.

FR-Z.1, Response to High Containment Pressure

This guideline is identical to its BASIC counterpart.

FR-Z.2, Response to Containment Flooding

This guideline is FR-Z.2 in BASIC, but renamed from "Response to High Containment Sump Level." It is identical to its BASIC counterpart.

FR-Z.3, Response to High Containment Radiation Level

This guideline is identical to its BASIC counterpart.

FR-I.1, Response to High Pressurizer Level

This guideline is FR-I.1 in BASIC, but renamed from "Response to Pressurizer Flooding." It is identical to its BASIC counterpart.

FR-I.2, Response to Low Pressurizer Level

This guideline is also FR-I.2 in BASIC, but renamed from "Response to Low System Inventory." It is identical to its BASIC counterpart.

FR-I.3, Response to Voids in Reactor Vessel

This guideline is nearly identical to its BASIC counterpart. RCP restart is now required before venting is allowed. We find this change acceptable.

4.0 CONCLUSION

The staff had previously reviewed the BASIC version of the ERGs (Reference 1) and reviewed Revision 1 of the ERGs for changes (Reference 7) from the BASIC version that were made in response to the staff's comments on the BASIC version. This report covers our review of the additional differences between the BASIC version and Revision 1. In discussing the generic issues and individual guidelines, this report provides amplification of some of the issues in our previous SER (Reference 7) and points out that with regard to Pressurized Thermal Shock and Stagnant Reactor Coolant Loops, the staff has not completed its review and therefore changes may be necessary to the related guidelines FR-P.1 and FR-P.2

Based on this review, we have concluded that Revision 1 of the ERGs is a substantial improvement over the BASIC version and that implementation of Revision 1 should continue while the staff's outstanding comments are resolved as part of the program for maintenance of the ERGs.

1. Safety Evaluation of Emergency Response Guidelines, June 1, 1983 (Generic Letter 83-22, June 3, 1983).

2. Letter, J. J. Sheppard (WOG) to D. G. Eisenhut (NRC), September 27, 1983, WOG Response to SE of WOG-ERGs.

3. Letter, J. J. Sheppard (WOG) to H. L. Thompson (NRC), November 30, 1983, Transmittal of ERG Revision 1.

4. Letter, J. J. Sheppard (WOG) to H. L. Thompson (NRC), May 4, 1984, Background for ERGs.

5. Letter, J. J. Sheppard (WOG) to D. G. Eisenhut (NRC), August 15, 1984, Comparison of Revision 1 ERGs with Basic.

6. Letter, D. G. Eisenhut (NRC) to J. J. Sheppard (WOG), December 27, 1984, Preliminary Review of ERG Revision 1.

7. Letter, T. M. Novak (NRC) to D. Butterfield, December 26, 1985, SSER on WOG ERGs.

8. NRC Generic Letter 85-12 to all Applicants and Licensees with Westinghouse Designed Nuclear Steam Supply Systems, June 28, 1985.

9. Letter, H. L. Thompson (NRC) to J. J. Sheppard (WOG), March 11, 1985, SSER on Basic version of ERGs.

10. Westinghouse Electric Corporation, A Generic Assessment of Significant Flaw Extension, Including Stagnant Loop Conditions, From Pressurized Thermal Shock of Reactor Vessels on Westinghouse Nuclear Power Plants, WCAP-10319, October 1983.

11. WCAP-10599, "Emergency Response Guidelines Validation Program Final Report."