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MEMORANDUM FOR: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Robert Minogue, Director
Office of Standards Development

Victor Stello, Director
Office of Inspection and Enforcement

FROM: Thomas E. Murley, Acting Director
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SUBJECT: RESEARCH INFORMATION LETTER # 98,
LIGHT WATER REACTOR STATUS MONITORING
DURING ACCIDENT CONDITIONS

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This memorandum transmits the results of completed research describing an improved method for analyzing accident sequences. The method is demonstrated by applying it to determine the operator's information needs during accidents. The results are relevant to the revision of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident." Appendix A summarizes the results and Appendix B is the detailed documentation on which this Research Information Letter is based.

1.0 Introduction

The accident at Three Mile Island in March 1979, and the results of subsequent investigations have reemphasized the importance of reactor operators and the role they play in determining the level of safety associated with nuclear power. At the same time, the adequacy of some longstanding regulatory approaches to safety, such as design basis events and the single failure criterion, is being questioned. Alternate methods, some employing insights from probabilistic risk assessment, are being proposed to broaden our perspectives on reactor safety.

This research introduces an analytical approach which could make significant contributions to accident analysis. As an illustration, the approach is used to identify the necessary and sufficient set of light water reactor instrumentation needed by analyzing the appropriate operator response to specific plant states associated with risk significant accident sequences. The resultant set of measurable parameters is compared to the list of such parameters in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs During and Following an Accident."

Criterion 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," includes a requirement that instrumentation be provided to monitor variables and systems over their anticipated ranges for accident conditions as appropriate to ensure adequate safety.

Criterion 19, "Control Room," of Appendix A to 10 CFR Part 50 includes a requirement that a control room be provided from which actions can be taken to maintain the nuclear power unit in a safe condition under accident conditions, including loss-of-coolant accidents, and that equipment, including the necessary instrumentaion, at appropriate locations outside the control room be provided with a design capability for prompt hot shutdown of the reactor.

Criterion 64, "Monitoring Radioactivity Releases," of Appendix A to 10 CFR Part 50 includes a requirement that means be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluid, effluent discharge paths, and the plant environs for radioactivity that may be released from postulated accidents.

Regulatory Guide 1.97 describes a method acceptable to the NRC staff for complying with the Commission's regulations to provide instrumentation to monitor plant variables and systems during and following an accident in a light-water-cooled nuclear power plant. The most recent version of the guide (Revision 2 dated June 1980) contains a list of variables to be measured together with the associated measurement range and purpose for the measurement. The design criteria (e.g., qualification and display requirements) for the associated instrumentation are also identified. This list was assembled by surveying the NRC staff and by reviewing accident response procedures involving preplanned manual actions during design basis events. Interactions among the staff, the Advisory Committee on Reactor Safeguards, licensees, applicants, vendors, and other interested members of the public have resulted in modifications to the original list. For the most part, it is a product of engineering judgment based on past experience and on the perceptions of individuals as to the significance of particular parameters and the impacts of implementation.

The research described herein developed a more systematic approach to determining instrumentation requirements. The application of the technique tends to confirm the reasonableness of the list generated via engineering judgment. It also identifies, however, differences whose significance should be reviewed by the staff.

2.0 Discussion

The analysis reported here is based on two observations concerning the enhancement of operator capabilities:

1. The operator's capability to diagnose and respond correctly to accident conditions is sensitive to the amount and quality of information available to him through the plant instrumentation. Accordingly, one of the primary objectives of this analysis was to determine systematically the necessary and sufficient set of plant instrumentation which would satisfy the operator's informational needs during accident conditions.

2. While there exist many diverse aspects of the general operator/plant interface problem, any efficacious changes to present designs and/or procedures must be based upon a foundation consisting of a thorough understanding of the plant response to accident events and a careful delineation of the specific responsibilities of the operator as the accident sequence progresses. Therefore, an additional objective of this analysis was to develop such a foundation upon which both this and additional analyses concerning enhanced operator capability could be performed.

The technical approach used in this analysis to accomplish the objectives outlined above was based on evaluating appropriate operator response in a logical progression of events. This approach can be succinctly summarized by addressing three fundamental questions.

1. What actions can (or must) the operator take in response to the accident condition?

2. What information is required by the operator to take this action?

3. What instrumentation is necessary and sufficient to provide this information?

By translating the general objectives into these three interrelated questions, the analysis could be performed systematically, increasing assurance that important operator informational needs will not be overlooked.

The approach is diagrammed in Figure 1. The seven accident sequences analyzed were determined to dominate risk in the previous risk analyses from which they were selected. All sequences involved system failures in excess of the single failure criterion.

For each sequence the physical response of the plant is defined in terms of measurable parameters. The time-dependent variations and the interrelationships of these parameters generate an "accident signature," a uniquely characteristic array which can be used to evaluate the status of the plant.

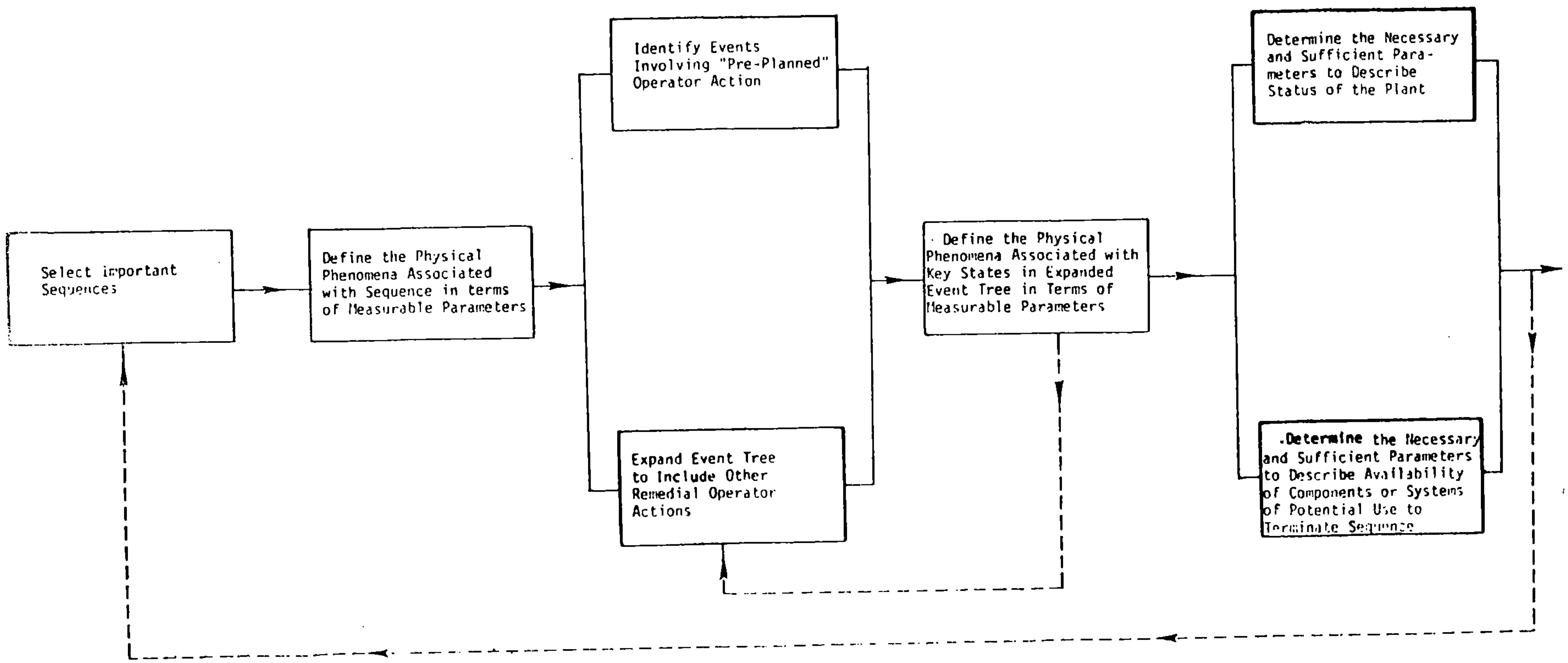

Figure 1. Flow chart of technical approach used to determine operator information needs and instrumentation requirements. Dotted lines imply possible feedback in an iterative process.

The development of the event trees began with the trees as they appeared in the original risk analyses. The events in each sequence which involved operator action were identified and in some cases broken down into additional events in order to highlight individual operator tasks. In addition, the sequences were expanded (events added to the event tree) to include additional operator actions which could be performed to prevent core melt, but were not taken credit for in the original analysis. These additional events included "repair events," where the operator is given the opportunity to attempt to restore or replace a particular function, and "delay events," where the operator is called upon to delay an inevitable melt as long as possible or to perform some other consequence mitigating action. The result of these efforts was an "operator action event tree" which identified success paths and which logically displayed the role of the operator throughout the progression of the accident. Figure 2 presents a simple example of such a tree developed for interfacing systems LOCA (V) sequence of WASH-1400.

Once the event logic and physical response of the plant are established, it is relatively straightforward to identify the key operator actions and the operator's information requirements. This is done by characterizing the status of the plant on each branch of the tree and associated appropriate actions in terms of physically measurable parameters. Table I summarizes this information for the V-sequence.

Prior to presenting the results, it is important to point out that this work represents a first-of-a-kind study conducted over a short time period. As such, there are limitations involved and refinements to be made in the analysis. These are delineated in Appendix B, Section 5.

3.0 Results

The results of this study pertinent to the revision of Regulatory Guide 1.97 are summarized in Appendix A. The table lists the variables derived from the analysis, indicates the significance of each, and identifies those not contained in the Revision 2 to Regulatory Guide 1.97. This study yields results which compare quite favorably with Revision 2, despite the major variations in technical approach. There are, however, some specific differences worth noting, especially PWR reactor vessel water level, containment sump water temperature, process parameters associated with the low pressure injection system, and positions of various valves.

Speaking more generally, this research introduces some important new concepts and technical approaches which, if properly developed and applied, could make significant contributions to accident analysis. It emphasizes the perceptions of the operator, the needs for information and the alternative successful actions one might take given various combinations of component failures. Beyond determining instrumentation requirements, the methods have important implications with respect to

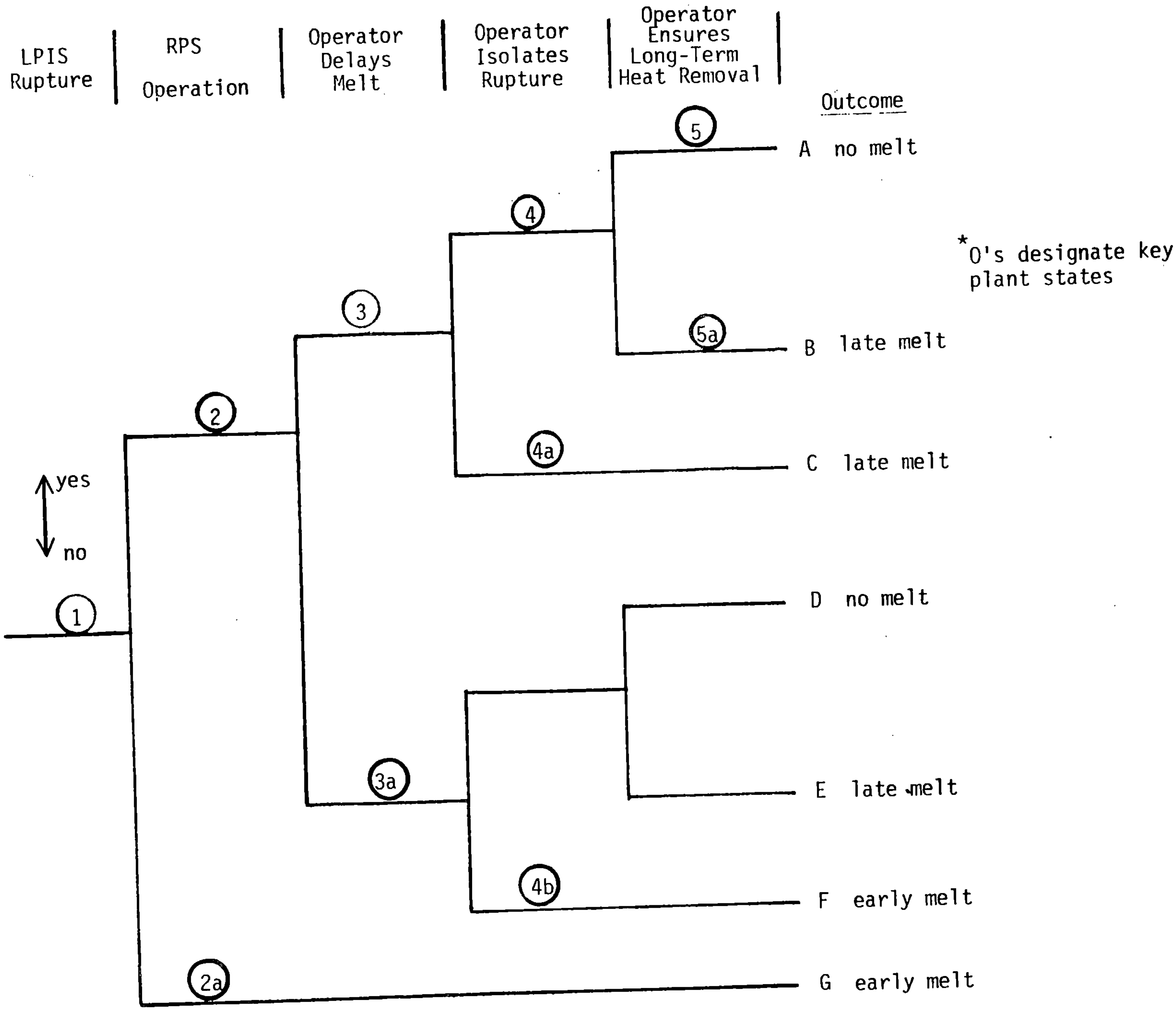

Figure 2. Interfacing Systems LOCA Operator Action Event Tree

Table I. Summary of Key Operator Actions and Information Requirements for V-Sequence

PLANT STATE (See Figure 2)	DESCRIPTION OF PLANT STATE	INFORMATION REQUIRED TO IDENTIFY PLANT STATE	APPROPRIATE OPERATOR ACTION FOLLOWING STATE IDENTIFICATION	INFORMATION REQUIRED TO TAKE APPROPRIATE ACTION
1	Rupture of check valves results in LPIS overpressure and rupture	• RCS P,T • Pressurizer water level • Containment P,T,R • Aux. Building T,R • LPIS P,T,R,F	Prepare for actions illustrated in Fig.4.8	See states 2, 3, 4 and 5
2	Reactor scram; decay power level; RCS pressure rapidly decreasing to HPIS actuation level	Control Rod Position Neutron flux	Initiate core melt delay actions and isolation	RCS P,T Vessel water level HPIS flow Accumulator flow Accumulator Tank level LPIS flow from RWST CSIS flow from RWST RWST level Isolation valve(s) position
2a	Reactor not scrammed; power level above capacity of HPIS to remove heat; core melt assumed to follow	Control Rod Position Neutron Flux RCS P,T	Monitor approach to cladding failure; initiate consequence mitigation systems	Primary system radiation level Aux. Building R
3	Minimum sufficient flow from HPIS to keep core covered and prevent melt	RCS P,T Vessel water level RWST level LPIS flow from RWST CSIS flow from RWST	Initiate (or continue) isolation actions	Isolation valve(s) position
3a	Either insufficient HPIS flow or excessive draw on RWST	Same as 3	Same as 3	Same as 3

Table I. (Continued)

PLANT STATE (See Figure 2)	DESCRIPTION OF PLANT STATE	INFORMATION REQUIRED TO IDENTIFY PLANT STATE	APPROPRIATE OPERATOR ACTION FOLLOWING STATE IDENTIFICATION	INFORMATION REQUIRED TO TAKE APPROPRIATE ACTION
4	LOCA successfully isolated before core melt occurs	Isolation valve position RCS P LPIS flow Pressurizer water level	Initiate long-term heat removal	RCS P,T Vessel water level Steam generator water level Auxiliary FW flow CST level Reactor power level
4a	Isolation fails after delaying action core melt occurs when RWST depleted	Same as 4	Monitor approach to core melt and initiate consequence mitigation actions	Primary system radiation level RWST level Aux. Building R
4b	Isolation fails; no delaying action has occurred; core melt occurs more quickly than 4a	Same as 4a	Same as 4a	Same as 4a
5	Long-term heat removal established	RCS P,T Steam gen. level Aux. FW flow		
5a	Long-term heat removal not established; no corrective action possible	RCS P,T Steam gen. level Aux. FW flow	Initiate consequence mitigation systems	

P = Pressure T = Temperature
R = Radiation Level F = Flow Rate

developing emergency procedures, generating training simulator exercises, and designing operational aids, including computerized diagnostic systems. Therefore, the methodology itself, as described previously in Section 2.0, should be viewed as a major result of this research.

4.0 Recommendations

The following recommendations are made with respect to the results reported here:

1. The regulatory and standards development staffs should review the concepts and technical approach described in Appendix B and advise the research staff as to the value and validity of these techniques, areas for their improvement, and suggested topics for their application. Assuming the methods are deemed promising, the regulatory staff may also want to encourage licensees and applicants to apply them to their own facilities.

2. The regulatory and standards development staffs should review these results and assimilate them into the technical basis for decisions relative to the revision of Regulatory Guide 1.97. Appropriate considerations should be given to the limitations of the study which generated these results.

In the meantime, RES is continuing this research. Additional accident sequences are being analyzed as is a broader spectrum of reactor designs. Furthermore, the development of best-estimate codes to calculate the physical response of plant systems during accidents continues to provide updated information on which to base these analyses.

The RES technical contact for this work is Raymond DiSalvo.

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Enclosures:
1. Appendix A: Summary of Variables Identified in Sequence Evaluations
2. Appendix B: LWR Status Monitoring During Accident Conditions (NUREG/CR-1440)

APPENDIX A

APPENDIX A - Summary of Variables Identified in Sequence Evaluations					
PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB′	
Control Rod Position	• Verification of scram	Same as V	Same as V	Same as V	Provides primary indication of successful scram
Neutron Flux	• Verification of scram	Same as V	Same as V	Same as V	Indicates shutdown margin; important after initial failure to scram; might be unreliable under voiding conditions
RCS Pressure	• Diagnosis of initiating LOCA event • Determination of need for and effectiveness of ECI • Provides, along with RCS temperature, degree of subcooling • Indication of break isolation	• Identification of initiating small break • Determination of need for and effectiveness of ECI and ECR • Provides, along with RCS temperature, degree of subcooling	Same as S_2C	• Indication of transient initiator • Indication of integrity of primary system • Provides, along with RCS temperature, degree of subcooling	
RCS Temperature	• Provides, along with RCS pressure, degree of subcooling	Same as V	Same as V	• Provides, along with RCS pressure, degree of subcooling • Indicator of natural circulation	Measurements of both hot and cold leg temperatures useful for natural circulation
Pressurizer Level	• Indication of initiating event • Indication of isolation of break	• Indication of initiating event • Diagnosis of size and location of break	Same as S_2C	• Indication of initiating event	

Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence: V	S_2C	S_1HF	TML/TMLB'	Comments
Pressurizer Relief Valve position, discharge line flow, or drain tank level				•Verification of pressurizer relief valve reclosure	Other parameters designed to indicate RCS integrity can be used as back-up to these direct indications
Vessel Water level	•Indication of need for and effectiveness of ECI •Indication of isolation of break	•Indication of initiating event •Indication of need for and effectiveness of ECI	Same as S_2C	•Indication of initiating event •Verification of relief valve closure and success of maintaining adequate liquid inventory	Not included in Reg. Guide 1.97. Other thermodynamic parameters (e.g. RCS pressure and temperature) can be used for most accident conditions. Further analysis is required to determine if these parameters are sufficient for all significant accident conditions
Primary System Radiation Level	•Indication of approach to core melt •Assessment of extent of core damage following restoration of core cooling	Same as V	Same as V	Same as V	On-line timely measurements are necessary; system should remain operable under all accident conditions including containment isolation
Boron Concentration	•Indication of shutdown margin	Same as V	Same as V	Same as V	Could be useful back-up if accident progresses to conditions which make neutron flux monitors unreliable

Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence: V	S_2C	S_1HF	TML/TMLB'	Comments
Containment Pressure	• Diagnosis of initiating LOCA	• Diagnosis of initiating break • Indication of CSIS failure, repair of CSIS, and effectiveness of CSRS • Provides, in combination with sump water temperature, indication of adequate NPSH for ECR pumps. • Indication of containment integrity	• Diagnosis of initiating break • Provides, in combination with sump water temperature, indication of adequate NPSH for ECR pumps • Indication of containment integrity • Indication of CSRS failure or effectiveness	• Verification of relief valve reclosure • Indication of containment integrity	
Containment Isolation Valve Position		• Verifies containment isolation to preclude transport of radioactive material through containment penetrations	Sames as S_2C	Same as S_2C	

Summary of Variables Identified in Sequence Evaluations					
PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence				Comments
	V	S_2C	S_1HF	TML/TMLB'	
Containment Temperature	•Diagnosis of initiating LOCA	•Diagnosis of initiating break •Indication of CSIS failure, repair of CSIS, or effectiveness of CSRS	•Diagnosis of initiating break •Indication of CSRS failure or effectiveness	•Verification of relief valve reclosure	Containment humidity can be used as a highly reliable backup to containment pressure and temperature to indicate primary system integrity
Containment Radiation Level	•Diagnosis of initiating LOCA	Same as V	Same as V		Serves as backup to containment pressure and temperature for indication of loss of primary boundary integrity
Containment Sump Water Level		•Indicate availability of water for ECR and CSRS	•Indicate absense of coolant flow between upper and lower compartment and successful restoration of flow		Can also be used as indicator of initiating break
Containment Sump Water Temperature		•In conjunction with containment pressure, indicates adequate NPSH for CSRS and ECR pump operation	Same as S_2C		Not included in Reg. Guide 1.97

Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence: V	S_2C	S_1HF	TML/TMLB'	Comments
Upper Containment Compartment Water Level and Drain Valve (between upper and lower compartments) position			• Indication of major cause for ECCS recirculation failure • Indication of repair and restoration of flow		Not specifically identified in Reg. Guide 1.97 but only applicable to plants with similarly designed containment drain system
Steam Generator Level	• Indication of capability of long term decay heat removal	• Indication of feedwater system performance	Same as S_2C	• Indication of initiating transient • Indication of performance of auxiliary system	
Steam Generator Pressure	• Indication of capability of long term decay heat removal	• Indication of feedwater system performance • Indication of secondary system integrity	Same as S_2C	• Indication of performance of feedwater system • Indication of capability of using condensate pumps (TML)	
Steam Generator Safety/Relief Valve Positions		• Indications of secondary system integrity	Same as S_2C	Same as S_2C	
Main Feedwater Flow				• Indication of initiator, success of repair, or utilization of condensate pumps (for TML)	Pump discharge pressure (not included on Reg. Guide 1.97) could be used as backup indication and assist in specifying cause of failure for TML

Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence: V	S_2C	S_1HF	TML/TMLB'	Comments
Auxiliary Feed-water Flow	• Indication of adequate water flow to steam generators for long term decay heat removal	• Indication of adequate flow to steam generators to enhance heat removal	Same as S_2C	• Indication of AFWS failure and determination of restoration	Pump discharge pressure could be used as backup; flow control valve positions could be useful in determining cause of AFWS failure and in regulation of restored AFWS
Condensate Pump Flow or Discharge Pressure				• Potentially useful in diagnosis of initiating event • Indication of effectiveness of using condensate pumps to supply feed-water to steam generators for some TML initiators	Not included in Reg. Guide 1.97
Steam Supply to AFW turbine driven pump				• Diagnosis of AFW failure cause and subsequent repair	Not included in Reg. Guide 1.97
Accumulator Tank level, flow rate, and/or isolation valve position	• Indicate injection after initiator		Same as V		Passive system; indirect indication of performance can be obtained from other parameters
Condensate Storage Tank Level	• Indication of ability to use AFW as heat removal system	Same as V	Same as V	Same as V	

Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence: V	S_2C	S_1HF	TML/TMLB'	Comments
Refueling Water Storage Tank Level	• Indication of availability of water for ECI • Determination of optimum use of RWST water supply in core melt delaying actions	• Indication of availability of water for ECI	Same as S_2C		
HPIS Flow	• Indicates success of ECI for core melt delay actions	• Verification of ECI operation following initiator	Same as S_2C		Pump discharge pressure can be used as backup indication of system operation
LPIS pressure, temperature, radiation level, and/or flow	• Diagnosis of initiating event (differentiate from other events with similar RCS response) • Indication of isolation of break • Determination of break location				LPIS pressure, temperature, and radiation level not included in Reg. Guide 1.97
LPIS Isolation valve position	• Indication of success of isolation				Not included in Reg. Guide 1.97
Containment Spray flow (including CSIS and CSRS)	• Indication of need to isolate system for delaying actions	• Indication of failure of CSIS and subsequent repair	• Indication of operation containment heat removal		Pump discharge pressure can be used as backup indication of system operation

Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence: V	S_2C	S_1HF	TML/TMLB′	Comments
RHR Flow	• Indication of system operation for long term heat removal	Same as V	Same as V	Same as V	Pump discharge pressure can be used as backup indication of system operation
Positions of key valves in safety related systems (HPIS, LPIS, CSIS, CSRS, CHRS, RHR)	• Indication of capability of systems to operate when called upon • Diagnosis of failure	Same as V	Same as V	Same as V	Not specifically included in Reg. Guide 1.97
Component Cooling Water Flow in CHRS heat exchangers		• Indication of effectiveness of containment cooling using CSRS	Same as S_2C		
Component Cooling Water Flow to RHRS Heat Exchanges	• Indication of effectiveness of long-term heat removal	Same as V	Same as V	Same as V	
Auxiliary Building Temperature or Radiation level	• Diagnosis of initiating event • Determination of successful isolation of break				Auxiliary Building Temperature : not included in Reg. Guide 1.97

Summary of Variables Identified in Sequence Evaluations

PWR Measured Variable	Major Purpose for Indicated PWR Accident Sequence: V	S_2C	S_1HF	TML/TMLB'	Comments
Containment auxiliary heat removal fan discharge flow			• Indication of the amount of containment cooling which is being performed and the requirements for CSRS		Only applicable to plants with such a system
Status of Class-1E power supplies to key safety system components	• Verification of safety system availability	Same as V	Same as V	• Indication of safety system availability • Diagnosis of cause for AFWS failure	
Status of Non-Class-1E Power Supplies	• Verification of available power source	Same as V	Same as V	• Indication of initiating event for TMLB' and determination of restoration	

Summary of Variables Identified in Sequence Evaluations

BWR Measured Variable	Major Purpose for Indicated BWR Accident Sequence: TC	COMMENTS
Control Rod Position	• Indication of failure of automatic scram, and success/failure of manual insertion of rods	
Neutron Flux	• Indication of failure to scram and determination of effect of manual shutdown actions	
RCS Pressure	• Determination of effect of delayed scram • Need for and effectiveness of HPCI • Effectiveness of long term cooling • Secondary indication of reactor shutdown	
RCS Temperature	• Indication of effectiveness of core cooling (in combination with RCS pressure)	Location of instruments not yet determined; core exit temperature (as listed in Reg. Guide 1.97) does not seem to be best location. Intended for those accident conditions where coolant level measurement might be expected to be unreliable
Vessel Water Level	• Indication of initiating transient event • Indication of water inventory • Determination of need for and effectiveness of emergency core cooling • Determination of when to secure HPIS and rely on RCIC for long term cooling	
Main Steam Flow Isolation Position	• Indication of initiator • Determination of potential core cooling procedures	MSIV should automatically close following the initiating loss of feedwater transient event

Summary of Variables Identified in Sequence Evaluations		
BWR Measured Variable	Major Purpose for Indicated BWR Accident Sequence TC	COMMENTS
Safety/Relief Valve Positions in Primary System (including ADS)	• Indication of effect of delayed shutdown • Indication of potential effectiveness of manual shutdown using SLCS • Indication of primary boundary integrity	
Radiation Level in Coolant	• Information for monitoring of core melt • Indication of amount of core damage	
Containment Pressure	• Indication of integrity of primary pressure boundary • Indication of containment integrity	
Containment Temperature	• Indication of integrity of primary pressure boundary • Indication of containment integrity	
Containment Radiation Level	• Indication of integrity of primary pressure boundary	
Suppression Pool Level	• Indication of primary coolant boundary integrity • Indication of availability of water for ECR	
Suppression Pool Temperature	• Indication of ability of cooling system to pump water	
Boron Tank Level	• Indication of Boron injection for shutdown	
SLCS flow or pump discharge pressure	• Indication of system operation	

Summary of Variables Identified in Sequence Evaluations

BWR Measured Variable	Major Purpose for Indicated BWR Accident Sequence: TC	COMMENTS
Boron Concentration	• Determination of effectiveness of manual shutdown using SLCS; indication of shutdown margin	Not included in Reg. Guide 1.97. Could be useful backup under accident conditions which make neutron flux monitors less reliable
Feedwater flow	• Indication of initiating event	
Feedwater pump discharge pressure current to pumps, or controller position	• Indication and diagnosis of cause of initiator	
RCIC valve positions	• Ensure availability of system	Not specifically included in Reg. Guide 1.97
Steam flow to RCIC turbine	• Indication of adequate flow to ensure system operation	
RCIC flow or pump discharge pressure	• Indication of successful system operation or cause of failure	
HPCS valve positions	• Ensure availability of system	Not specifically included in Reg. Guide 1.97
HPCS flow, pump discharge pressure, or current to pumps	• Indication of successful system operation or cause of failure	

Summary of Variables Identified in Sequence Evaluations

BWR Measured Variable	Major Purpose for Indicated BWR Accident Sequence TC	COMMENTS
RHR valve position (valves required for pre-warming and flushing and flow control valves)	• Allow startup of system and subsequent operator control of flow	Not included in Reg. Guide 1.97
RHR heat exchanger inlet/outlet temperature	• Information necessary for manual startup and indication of subsequent system performance	
HPSW valve position	• Indication of availability of system	
HPSW flow or pump discharge pressure	• Indication of system operation	