Consumers
Power
Company

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July 31, 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Director, Nuclear Reactor Regulation
Att Mr Dennis M Crutchfield, Chief
Operating Reactors Branch No 5
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT -
EVALUATION OF ADDITIONAL WIDE-RANGE LEVEL INSTRUMENTATION
USING PRA BASED METHODS; RESPONSE TO NUREG-0737 ITEM II.F.2

Consumers Power Company submittal dated July 9, 1981 reaffirmed our commitment to continue evaluating the usefulness of wide-range level instrumentation as part of the Probabilistic Risk Assessment (PRA) based Continuing Risk Management Program. This evaluation was accomplished by developing and analyzing Operator Action Event Trees. As committed to in our submittal of July 9, 1981, this letter provides Consumers Power Company's final evaluation of the usefulness of augmenting the existing instrumentation at the Big Rock Point Plant with wide-range level instrumentation.

A summary of our evaluation, including a description of the development of Operator Action Event Trees and the results of analyzing these trees is provided in Attachment 1. Supplementary information used in this effort is also provided in Attachments 2 through 5. As a result of this evaluation, Consumers Power Company has concluded that existing instrumentation at Big Rock Point provides unambiguous indication of inadequate core cooling as well as the approach of inadequate core cooling. Additional wide-range level instrumentation would provide only indirect indication of failures in core cooling systems and would serve as a redundant backup to the existing instrumentation.

Based on these conclusions, Consumers Power Company does not intend to install wide-range level instrumentation at the Big Rock Point Plant. Accordingly, Consumers Power Company will not be submitting a plan of action for installing additional instrumentation by September 1, 1981. This letter completes our response to NUREG-0737 Item II.F.2.

David P Hoffman

David P Hoffman
Nuclear Licensing Administrator

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CC Director, Region III, USNRC
NRC Resident Inspector - Big Rock Point

CONSUMERS POWER COMPANY
Big Rock Point Plant

NUREG-0737, Clarification of TMI Action Plan Requirements
Supplemental Response to NRC Letter, dated October 31, 1980

Docket No 50-155
License No DPR-6

At the request of the Commission and pursuant to the Atomic Energy Act of 1954, and the Energy Reorganization Act of 1974, as amended, and the Commission's Rules and Regulations thereunder, Consumers Power Company submits our supplemental response to NRC letter dated October 31, 1980 (NUREG-0737-"Clarification of TMI Action Plan Requirements", Item II.F.2). Consumers Power Company's supplemental response, entitled "Evaluation of Additional Wide-Range Level Instrumentation Using PRA-Based Methods; Response to NUREG-0737 Item II.F.2", is dated July 31, 1981 and provides information in addition to that provided by our responses dated December 1979, 1980 and July 9, 1981.

CONSUMERS POWER COMPANY

By ______________________
J W Reynolds, Executive Vice President

Sworn and subscribed to before me this 31st day of July 1981.

Helen I Dempski
Helen I Dempski, Notary Public
Jackson County, Michigan
My commission expires December 14, 1983

ATTACHMENT 1

Summary of Wide-Range Level Instrumentation Evaluation

Introduction

Item II.F.2 of NUREG-0737 presented the following NRC position with respect to instrumentation for detection of inadequate core cooling.

"Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided."

Expanding on this position, the NRC further indicated that this instrumentation must also indicate the approach of inadequate cooling, the existance of inadequate cooling from multiple causes, must not erroneously indicate inadequate cooling due to unrelated phenomena, and must indicate full range from normal operating conditions to full core uncovery. Evaluation of this instrumentation was to include reactor level indication.

In our initial response to NUREG-0737, dated December 19, 1981, Consumers Power indicated its participation in the work being performed for the General Electric Owners Group with respect to the need for wide-range level indication. The result of this work was that the existance of adequate core cooling could be verified by assuring that existing vessel water instrumentation indicated water level above the core or that rated flow was indicated in one of the low pressure core spray loops. The conclusions of the Owners Group were felt to be applicable at Big Rock Point. Consumers Power also indicated that procedures requiring use of this instrumentation for the detection of inadequate cooling would be implemented as dictated by the schedule of Item I.C.1 of NUREG-0737 (Guidance for Development of Emergency Procedures). Any further consideration of the addition of wide-range level instrumentation would occur pending the completion of the Probabilistic Risk Assessment (PRA) of the Big Rock Point Plant.

Consumers Power Company PRA submittal, dated March 31, 1981, reported that the need for wide-range level instrumentation was not indicated in any of the dominant accident sequences. Addition of this instrumentation would not result in an improvement of the safety of operation of the Big Rock Point Plant, and continued deferral of this requirement was in order. It was proposed that as a part of the Continuing Risk Management Program at Big Rock, an explicit examination of the benefits of wide-range instrumentation would be performed by expanding key accident sequences as defined in the PRA into operator action event trees using the methodology of NUREG-CR/1440 (see Attachment 2). To assure that the operator could perform critical functions during accident sequences, the specific needs of the operator with respect to instrumentation could be identified using this methodology. If wide-range level instrumentation proved useful as a result of this evaluation, appropriate instrumentation would be identified and modification proposed.

Development of Operator Action Event Trees

Examination of the benefits of wide-range level instrumentation using the approach defined in NUREG-CR/1440 has been completed. Operator Action Event Trees for key dominant accident sequences were developed and the instrumentation required to ascertain the existance of inadequate core cooling identified. Selection of accident sequences for this study was based on the following criteria: 1) the sequences result in core uncovery with the potential for significant core damage unless operator action is taken, 2) the sequences have the potential for causing ambiguous information to be transmitted to the operator with respect to action necessary to respond to the accident, 3) the sequences must be dominant from a probabilistic standpoint. Attachment 3 identifies and briefly describes the selected sequences.

These sequences include small break LOCAs below the core, large break LOCAs above and below the core which coincidentally disable portions of, or both of the core spray lines to the vessel, spurious blowdown of the primary system through the turbine bypass line, loss of instrument air, loss of the main condenser, loss of off-site power including loss of all AC power sources, and ATWS. The selected sequences encompass a wide spectrum of potential transients and accidents at Big Rock Point.

Equally important, the Operator Action Event Tree branches and operator actions discussed here are representative of transient and accident sequences other than those apecifically developed for this study. Consideration was given to the identification of inadequate core cooling for both long term decay heat removal situations as well as short term core cooling due to core uncovery.

At each of the branch points developed for the operator action event trees, three items were specifically considered; 1) what the actual level of the primary system was at that branch point, 2) whether the operator knew what this level was, and 3) what additional information or benefits wide range level instrumentation into the core region would provide.

Results

With few exceptions, a typical sequence of operator actions occurs in all accident sequences depending on the information available from existing steam drum and reactor level indication (see Attachment 4):

- Reactor water level indication above Reactor Depressurization System (RDS) actuation setpoint (reactor level > 2'9" above core). This is indication that the core is covered and adequate cooling exists.

  Operator Action -

  1. Steam drum level below centerline or downscale. Add water via condensate/feedwater, control rod drive, or, if reactor is depressurized, core spray.

  2. Steam drum level at or above centerline. Control water injection as necessary via condensate/feedwater, control rod drive, or core spray systems (if reactor depressurized) to maintain drum level instrumentation on scale.

- Reactor water level indication below RDS actuation setpoint ($<$ 2"9" above core). This is indication that uncovering of the core is imminent or has occurred.

Operator Action -

1. Insure RDS actuation
   a. Check depressurization valves open via position indication, check reactor pressure low, and/or listen for noise associated with rapid depressurization from containment.
   b. If unsure of depressurization, manually actuate RDS.

2. Insure adequate core spray
   a. Check core spray flow indicators for sufficient flow.
   b. Check fire pumps (annunciated in control room) on and core spray valves open via position indication.
   c. If unsure of adequate flow, open redundant flow path to core spray system through post incident systems (MO7072).

3. Start condensate/feedwater system to provide contents of hotwell and condensate storage to the primary coolant system.
   a. Check feedwater flow for water addition.
   b. Check condensate pump current, feed pump current, feedwater control valve position, and hotwell level as backup to feedwater flow.

The benefits of wide-range level instrumentation into the core region are available when reactor level is low, that is when reactor level drops below the RDS actuation setpoint. The actions identified above to maintain vessel level are the only actions available for the operator to take when reactor inventory drops to this level. The installation of wide-range level instrumentation provides no additional indication that these actions need to be performed nor does it make available any additional systems to provide makeup to the primary coolant system.

Indication that there is inadequate cooling is ascertained by the concurrent indication of reactor water below the RDS actuation setpoint and insufficient core spray flow. Additional wide-range level indication can serve as backup to existing level instrumentation and provide indirect indication of inadequate spray flow in that vessel level is not recovering or recovering slowly. However, this additional instrumentation does not provide direct indication of adequate cooling nor does it provide any additional information with respect to the reason inadequate cooling is occurring.

The most beneficial aspect of wide-range level instrumentation is to provide one additional piece of information to the operator that indicates core spray and makeup systems are not working and, even though he has performed all appropriate actions, additional trouble shooting is required. Attachment 5 identifies such accident sequences in which core region level indication provides this information when it would not be available otherwise. The nature and number of failures required to result in such sequences are such that they can be considered probabilistically insignificant.

Conclusions

Existing instrumentation at Big Rock Point provides unambiguous indication of inadequate core cooling and its approach when it occurs. This instrumentation includes reactor vessel level indication and core spray flow. The operator can perform only a few specific actions if this instrumentation indicated the potential for inadequate cooling and procedures require him to perform them any time low reactor water level occurs. This conclusion is consistent with the BWR Owners Group position.

Additional wide-range level instrumentation provides indirect indication of failures in core cooling systems and can serve as redundant backup to existing instrumentation. Wide-range level indication provides no additional useful information to the operator in trouble shooting the cause of inadequate cooling. Additional wide-range level indication does provide some benefit to the operator in defining the best course of action for the operator to take during certain transient sequences which have been determined to be insignificant from a probabilistic standpoint.

Based on this study, Consumers Power Company concludes that little benefit can be derived from the addition of wide-range level instrumentation in the Big Rock Point primary coolant system and plans no further work activity in this area.

ATTACHMENT 2

Description of Operator Action Event Tree Methodology

NUREG/CR-1440 entitled "Light Water Reactor Status Monitoring During Accident Conditions" presents a systematic approach for addressing operator informational needs during reactor accidents. Particular Accident Sequences are obtained or developed in the form of event trees as were derived for the Reactor Safety Study (WASH-1400). The selected sequences are generally those which are significant from a risk standpoint.

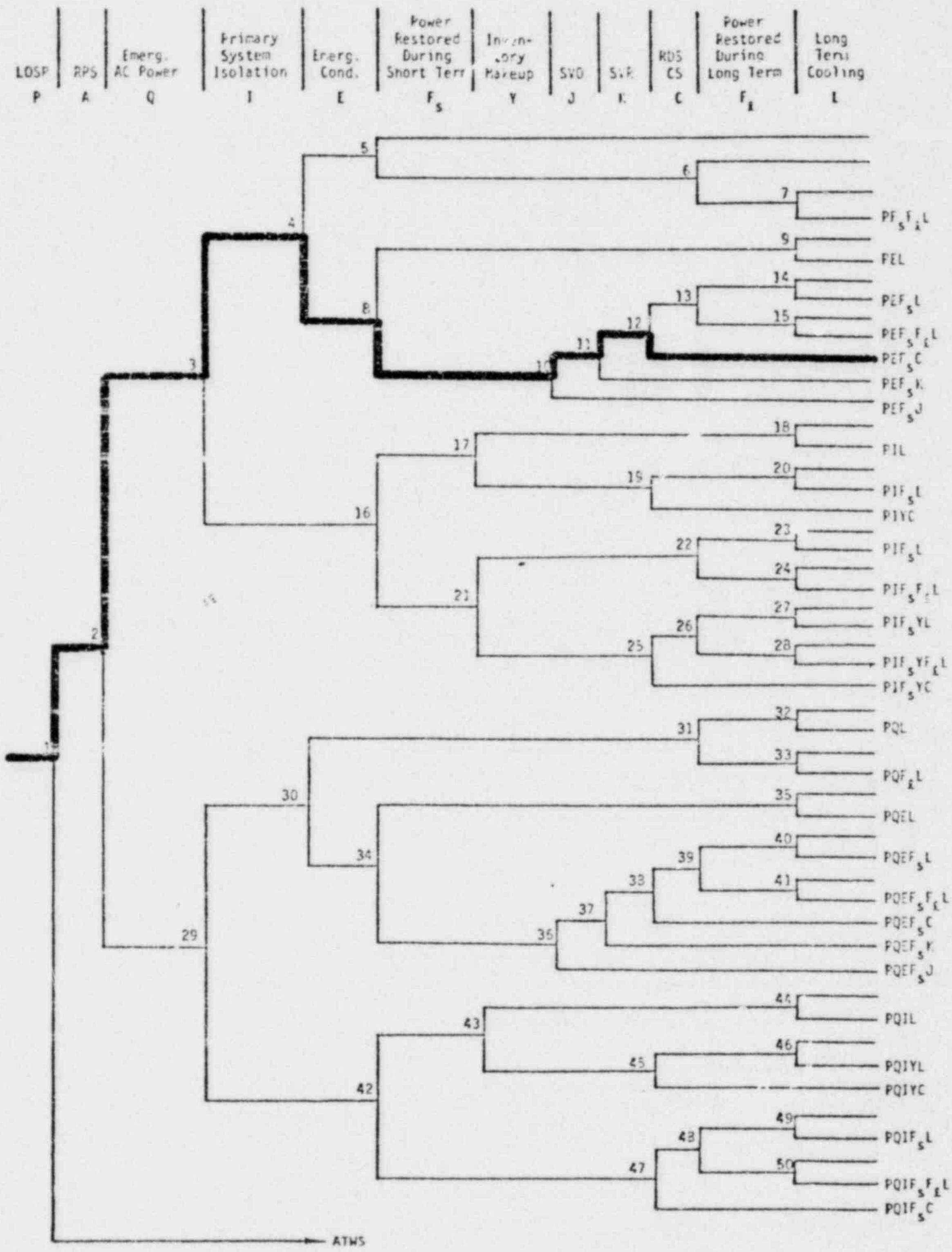
The heading of each branch of a particular event tree sequence identifies a unique plant state or equipment function. Operator action event tree methodology begins by identifying a set of reactor conditions for the plant states associated with the event tree headings. An appropriate operator response to this set of conditions is identified. These operator responses make up the headings of the operator action event tree for the sequence of interest.

With knowledge of the plant state and appropriate operator response, the instrumentation required to supply the operator with the necessary and sufficient information to respond appropriately can be identified. Success or failure of the operator to act is dependent on the availability of this instrumentation.

The source of the dominant accident sequences for studying the benefits of wide-range level instrumentation was Appendix I of the Big Rock Point PRA. Selected sequences for the wide-range instrumentation study are identified in Attachment 3. An example event tree and operator action event tree for the PE_VF_SC sequence follows.

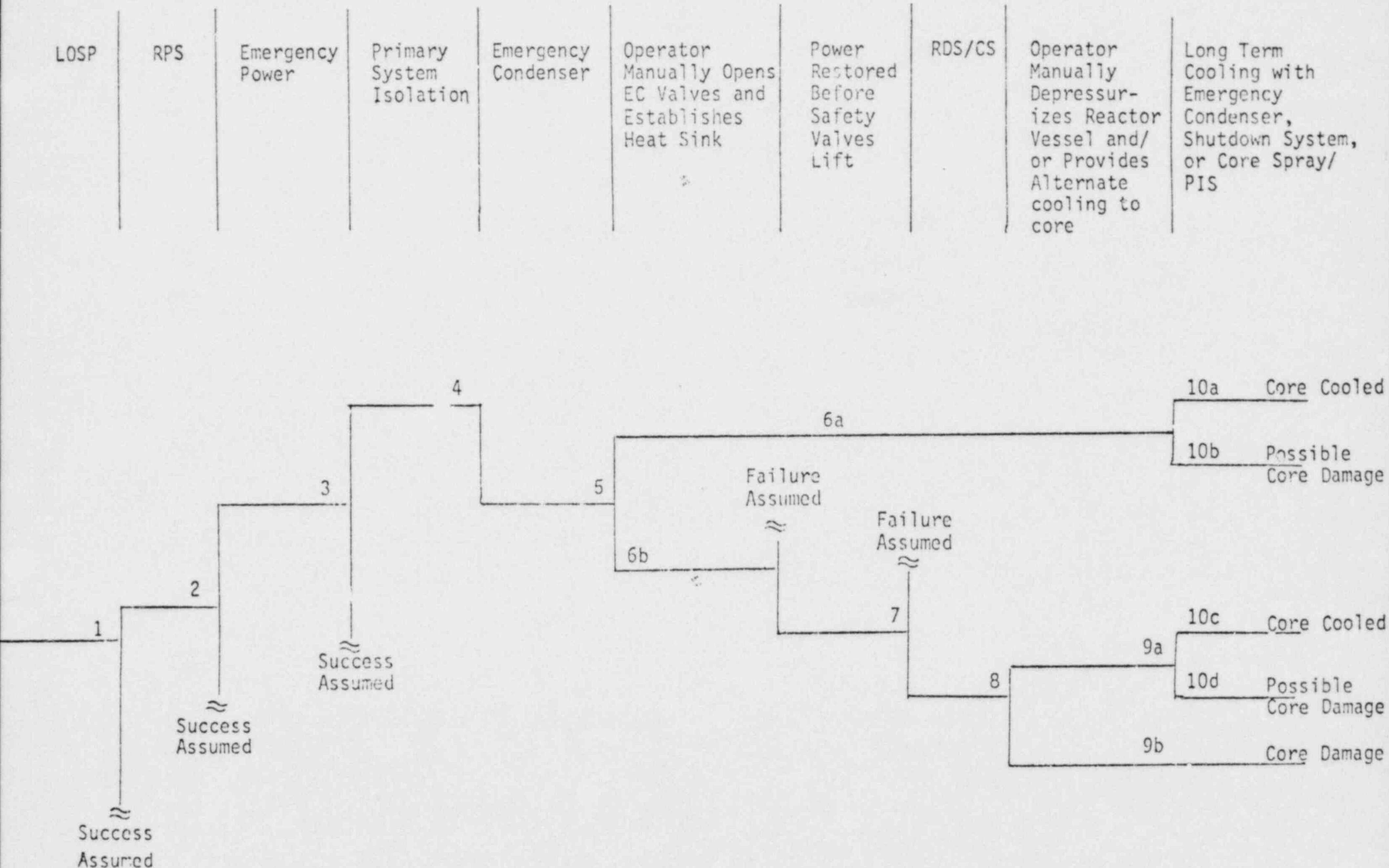
Attachment 2 continued

LOSS OF OFFSITE POWER EVENT TREE



Attachment 2 continued

OPERATOR ACTION EVENT TREE FOR THE PE_VF_SC SEQUENCE



ATTACHMENT 3

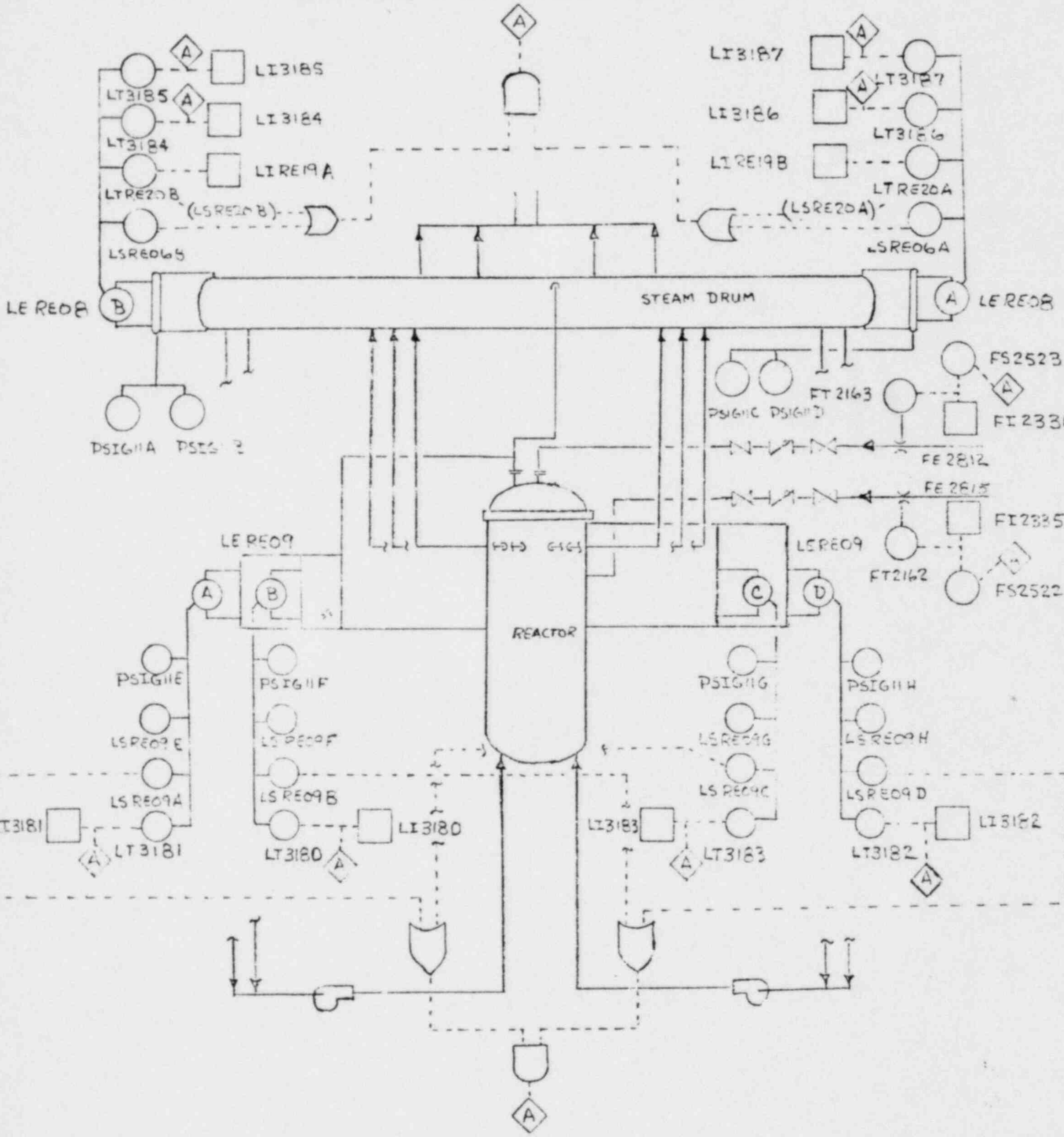
Wide-Range Level Instrumentation - Operator Action Event Trees

Sequence[1]	Probability[1] (yr^{-1})	Description
H_1Z	1.1×10^{-6}	High energy line break in the pipe tunnel outside containment which disables both core spray lines.
H_2	3.9×10^{-7}	High energy line break in the recirc pump room which disables both core spray lines or a portion of the core spray system.
BB_cZY_fC	2.0×10^{-5}	Spurious opening of the turbine bypass valve, failure to isolate the bypass line or MSIV, failure to makeup with feedwater, followed by RDS/core spray failure.
TAY_fL_r	1.4×10^{-6}	ATWS with loss of feedwater.
PE_vF_sC	3.1×10^{-6}	Loss of off-site power, failure of emergency condenser outlet valves, failure to restore off-site power quickly, RDS/core spray failure.
PQE_mF_sC	2.5×10^{-6}	Loss of off-site power, failure of diesel generator to start, failure to makeup to emergency condenser, failure to restore off-site power quickly, RDS/core spray failure.
ME_vNL	1.7×10^{-6}	Loss of main condenser, failure of emergency condenser outlet valves, failure to restore main condenser, long term cooling failure via post incident system.
UE_mUL	1.9×10^{-5}	Loss of instrument air, failure to makeup to emergency condenser shell, failure to restore instrument air, long term cooling failure via post incident system.
S_1E_mC	4.0×10^{-6}	Small break LOCA below the core, RDS/core spray failure.
S_1E_mL	3.7×10^{-5}	Small break LOCA below the core, long term cooling failure via post incident system.

(1) Sequences and probabilities from Appendix I of Big Rock Point PRA.

ATTACHMENT 4

Identification of Existing Instrumentation at Big Rock Point used to Indicate Inadequate Core Cooling



INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING BIG ROCK POINT

All Flow Indicators, Level Indicators and Annunciators Located in Control Room

Instrument	Description/Function	Range/Setpoint
FE2812	Redundant Core Spray Flow	600" H_2O
FT2163		0-600" H_2O
FI2336		0-900 gpm
FS2523		>100 gpm annunciates
FS2523		>500 gpm
FE2815	Primary Core Spray Flow	600" H_2O
FT2162		0-600" H_2O
FI2335		0-900 gpm
FS2522		>100 gpm annunciates
		>500 gpm
LERE08A	Steam Drum Level East	60" H_2O
LTRE20A		60" H_2O
LIRE19A		-30" to +30" St drum ℄
LT3186	RDS actuation	0-39.1" H_2O annunc. 17" below ℄
LI3386		-30" to +30" St drum ℄
LT3187	RDS actuation	0-39.1" H_2O annunc. 17" below ℄
LI3387		-30" to +30" St drum ℄
LERE08B	Steam Drum Level West	60" H_2O
LTRE20B		60" H_2O
LIRE19B		-30" to +30" St drum ℄
LT3184	RDS actuation	0-39.1" H_2O annunc. 17" below ℄
LI3334		-30" to +30" St drum ℄
LT3185	RDS actuation	0-39.1" H_2O annunc. 17" below ℄
LI3335		-30" to +30" St Drum ℄
LERE09A	Reactor Water Level	18" H_2O
LSRE09A	ECCS/RPS actuation	33" above Core
LSRE09E	B.U. ECCS actuation	33" above Core
LT3181	RDS actuation	0-27.4" H_2O annunc. 33" above Core
LI3381		24" to 42" above Core

Instrument	Description/Function	Range/Setpoint
LERE09B	Reactor Water Level	18" H_2O
LSRE09B	ECCS/RPS actuation	33" above Core
LSRE09F	B.U. ECCS actuation	33" above Core
LT3180	RDS actuation	0-27.4" H_2O annunc. 33" above Core
LI3380		24" to 42" above Core
LERE09C	Reactor Water Level	18" H_2O
LSRE09C	ECCS/RPS actuation	33"
LSRE09G	U. ECCS actuation	33" above Core
LT3183	DS actuation	0-22.4" H_2O annunc. 33" above Core
LI3383		24" to 42" above Core
LERE09D	Reactor Water Level	18" H_2O
LSRE09D	ECCS/RPS actuation	33" above Core
LSRE09G	B.U. ECCS actuation	33" above Core
LT3182	RDS actuation	0-27.4" H_2O annunc. 33" above Core
LI3382		24" to 42" above Core

Abbreviations:

ECCS - Emergency Core Cooling System

RPS - Reactor Protection System

B.U. - Back Up

RDS - Reactor Depressurization System

ATTACHMENT 5

Discussion of Accident and Transient Sequences in which Wide-Range Level Instrumentation May be Beneficial

1. Loss of Off-Site Power, no AC power (Emergency Diesel Generator, EDG, failure)

   During loss of off-site power, core spray flow indication derives its power from the emergency bus. Failure of on-site emergency AC power equipment will therefore disable this important indication of inadequate core cooling. To place the reactor in this situation in which inadequate cooling results, all of the following must occur:

   a. LOSP - loss of off-site power
   b. Emergency diesel failure to start or run
   c. Failure of the emergency condenser as a heat sink either by failure of the outlet valves to open or failure of makeup to the shell.
   d. Failure to repair the EDG by the time RDS actuation setpoint is reached.
   e. Failure to place the standby diesel on the emergency bus by the time the RDS actuation setpoint is reached
   f. Failure to restore off-site power by the time the RDS actuation setpoint is reached
   g. Failure of the RDS/core spray systems for reasons not noticeable from the control room (ie, other than core spray valves failure to open or diesel fire pump failure to start)

Probabilities of occurrence of items a through g are found in Appendices I, II, and III of the PRA. The total probability of occurrence of sequences involving loss of all AC power sources and the RDS/CS system (in a manner not noticeable from the control room) is estimated to be less than 1.6×10^{-7}/yr.

2. Sequences involving failure of RDS/CS and condensate/feedwater systems

   On recognition of failure of the RDS/core spray systems, the operator will begin injection of water to the primary system with the condensate/feedwater systems. Condensate/feedwater can be used in an attempt to reflood the vessel for all sequences in which off-site power is available and feedwater has not been disabled. Primary indication of feedwater addition to the primary system is feedwater flow. Backup indication includes condensate and feed pump motor currents, hotwell level, and feedwater reg valve position indication. From Appendix I of the PRA, dominant accident. sequences ending in RDS/core spray failure which could potentially be mitigated by rapid addition of feedwater occur with a frequency of 5.5×10^{-5}/yr.

   Wide-range level instrumentation would serve as backup to feedwater system indication for these sequences in that it verifies the water addition to the vessel. A lack of rising water level during feedwater addition indicates to the operator that the system is not adequate to keep up with primary coolant losses or additional trouble shooting of the feedwater system is required to satisfactorily reflood the core. The most likely cause of unsuccessful reflood of the vessel with feedwater while all feedwater instrumentation indicate normal is diversion of water through a rupture in piping downstream of the feedwater flow instrumentation. The likelihood of rupture of this particular section of piping simultaneous with the failure of RDS/core spray reduces the probability of these sequences to much less than 10^{-8}/yr.

3. Partial core spray failure as a result of a high energy line break below the core

Appendix I of the PRA identifies high energy line breaks within the containment which could potentially disable portions of the core spray system piping. The probability of occurrence of these sequences is 3.9×10^{-7}/yr.

Assuming these sequences are a result of breaks below the core and result in degraded core spray flow to the vessel, wide-range level instrumentation could be beneficial in assisting the operator in determining whether to keep the core cooled with core spray or feedwater. A range of break sizes exist in which feedwater is capable of keeping up with break flow and maintaining the core at least partially covered, but for which the degraded core spray system is unable. Wide-range level instrumentation would indicate that the level in the vessel would fall after feedwater tripped due to a lack of inventory in the hotwell. With the information that prior to the feedwater trip vessel level was near satisfactory, the operator could preferentially route fire system flow to the hotwell and maintain vessel level with this high volume system rather than attempting to cool with a degraded spray system.

The range of break sizes which fall in this category are believed to be in the 400-1000 gpm range. This is a very narrow range of breaks and comprises only a small fraction of the spectrum of medium and large breaks which make up to 3.9×10^{-7}/yr. high energy line break sequences.

The probability of occurrence of those LOCA sequences which would potentially benefit from wide-range level indication is considered to be much less than 10^{-8}/yr.