UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the matter of )

General Electric Company ) Docket No. STN 50-447

Standard Plant )

AMENDMENT NO. 10 TO APPLICATION
FOR REVIEW OF 238 NUCLEAR ISLAND GENERAL ELECTRIC
STANDARD SAFETY ANALYSIS REPORT (GESSAR II)

General Electric Company, applicant in the above captioned proceeding, hereby files Amendment No. 10 to the 238 Nuclear Island General Electric Standard Safety Analysis Report (GESSAR II).

Amendment No. 10 consists of two parts, a non-proprietary portion and a portion considered by the General Electric Company to be proprietary. The pages considered to be proprietary are so marked and are transmitted under separate cover.

Amendment No. 10 further amends GESSAR II by furnishing Appendix 1D; Assessment of Regulatory Guide 1.97, Revision 2 against GESSAR II Design.

Respectfully submitted,

General Electric

by: s/G. G. Sherwood
Glenn G. Sherwood, Manager
Nuclear Safety & Licensing Operation

STATE OF CALIFORNIA )
COUNTY OF SANTA CLARA) ss:

On this 23 day of November in the year 1982, before me, Karen S. Vogelhuber, Notary Public, personally appeared Glenn G. Sherwood, personally proved to me on the basis of satisfactory evidence to be the person whose name is subscribed to this instrument, and acknowledged that he executed it.

s/K. S. Vogelhuber
NOTARY PUBLIC, STATE OF CALIFORNIA
Santa Clara County
My Commission Expires December 21, 1984
175 Curtner Avenue
San Jose, CA 95125

csc/I110814-1

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INSTRUCTIONS FOR FILING AMENDMENT NO. 10

A tab is also included for Appendix 1D.

Remove and insert the pages listed below. Dashes (----) in the remove or insert column indicate no action required.

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# APPENDIX 1D

## ASSESSMENT OF REGULATORY GUIDE 1.97, REVISION 2 AGAINST GESSAR II DESIGN

# APPENDIX 1D

## ASSESSMENT OF REGULATORY GUIDE 1.97, REVISION 2 AGAINST GESSAR II DESIGN

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APPENDIX 1D

ASSESSMENT OF
REGULATORY GUIDE 1.97, REVISION 2
AGAINST GESSAR II DESIGN

## 1D.1 SUMMARY

This report provides an assessment of the 238 Nuclear Island design described by GESSAR II against the guidelines of Regulatory Guide 1.97, Revision 2, dated December 1980, "Instrumentation for Light Water Cooling Nuclear Power Plant to Assess Plant and Environmental Conditions During and Following an Accident." Any modifications to GESSAR required by implementation of changes identified by this assessment are under review.

### 1D.1.1 Introduction

Regulatory Guide 1.97 describes a method acceptable to the NRC staff for compliance with the Commission's regulation to provide instrumentation to monitor plant variables in systems during and following an accident at light water cooled nuclear power plants. Although the intent of the Regulatory Guide has been met, several exceptions have been made to the Regulatory Guide as written. These exceptions and their bases are discussed further in this report. These exceptions are viewed as acceptable means to implement the guide's intent.

All variables identified by the regulatory guide are identified in this report. Table 1D-1 summarizes these variables. However, since some variables are outside of the 238 Nuclear Island scope, they are left for the Applicant to assess.

### 1D.1.1 Introduction (Continued)

Since there are other modifications being made to address post-TMI changes (Appendix 1A), the assessment makes use of the existence of these changes and does not repeat a description of them here.

### 1D.1.2 Objective

This assessment is conducted in three stages. The first stage develops an implementation position which defines the specific requirements against which each variable is assessed. The implementation position is included as Attachment A and is discussed in Subsection 1D.2.1.

The second stage of the assessment defines the Type A variables. These variables are identified from a review of design basis accidents in Chapter 15 and the Emergency Procedure Guidelines. A more detailed discussion of the Type A variables is given in Subsection 1D.2.2.

The third stage of the assessment is a review of each variable shown on Table 1D-1. For this review, statements are provided to either justify the current design or to provide a conceptual definition of changes necessary to comply with the regulatory guide. These discussions may be found in Subsection 1D.2.3.

### 1D.1.3 Conclusions

Table 1D-2 summarizes the results of the assessment. The degree of compliance is shown by the information in the table and the accompanying remarks.

Table 1D-2 shows that the 238 Nuclear Island, as modified by the recommendations under review as a result of this assessment, fully complies with the intent of Regulatory Guide 1.97, Revision 2.

## 1D.2 ASSESSMENT

### 1D.2.1 Implementation Position

The implementation position defines the requirements of the regulatory guide as modified by GE to represent an acceptable means to implement the intent of the regulatory guide. The implementation document is included as Attachment A.

The implementation position is organized in the same manner as Section C of the regulatory guide. Where other documents have been referenced in the guide, the reference material is included instead. Where an exception or modification to the guide has been made, the information has been included in brackets. Discussion of these exceptions is included with a discussion of the variable in Subsection 1D.2.3.

The interpretation of Type A variables has been expanded to acknowledge variables considered by the Emergency Procedure Guidelines (Reference 1). Emergency actions, such as manual scrams or depressurization, were included in the review to identify Type A variables.

Clarification to the quality assurance requirements was added to reference the General Electric Quality Assurance Program.

### 1D.2.2 Type A Variable Assessment

Type A variables are fundamentally plant parameters needed to alert the control room operators to take safety actions by manually initiating a system or function which otherwise would not be automatically initiated in the course of an event. The regulatory guide does not specify Type A variables; rather it requires that each plant develop its own list of Type A variables from a review of each plant design.

For this assessment, the list of Type A variables was identified from a review of accidents described in Chapter 15 and a review of the Emergency Procedure Guidelines (Reference 1). The event descriptions of Chapter 15 and the Plant Nuclear Safety Operational

1D.2.2 Type A Variable Assessment (Continued)

Analysis (NSOA) of Appendix 15A were reviewed to determine the plant systems which require manual initiation and the key variables associated with manual initiation of those systems. The Emergency Procedure Guidelines prepared by General Electric for the BWR Owners' Group were also reviewed to identify any other variables requiring safety action and not identified by the review of the FSAR. A summary of the Type A variables are identified through this process as shown in Table 1D-3. Details of the Type A variable assessment are provided in Attachment B.

1D.2.3 Variable Assessments

This section summarizes the results of the individual variable assessments concentrating on deviations identified between the existing design of the 238 Nuclear Island and the implementation position for the regulatory guide.

Strict compliance with the regulatory guide is not recommended in all cases. In some cases, an acceptable alternate has been proposed which meets the intent to have meaningful post-accident indications. For some parameters, this can be met by adopting alternate variables to those specified in the regulatory guide or by specifying combinations of other variables. Another approach taken is to take exception to the guide where a reasonable justification can be provided. Finally, where a design change is indicated, the basic conceptual design is described as under review. When these reviews are completed, the resolution will be included in future amendments.

Many of the variables assessed do not meet the qualification requirements specified by the regulatory guide for Category 1 and Category 2 instruments. These requirements specify that the instrumentation "be qualified in accordance with Regulatory Guide 1.89... and the methodology described in NUREG-0588" for the

1D.2.3 Variable Assessments (Continued)

environmental qualifications of safety-related electrical equipment. In addition, the Regulatory Guide requires that "the seismic portion of qualification should be in accordance with Regulatory Guide 1.100."

The equipment requiring qualification is summarized in Tables 3.10-1 [MPL] (for seismic qualification) and Tables 3.11-9 [MPL] (for Environmental Qualification). The environmental qualification is to be in accordance with IEEE 323-1974 methods and the envelopes summarized in Tables 3.11-2 through 3.11-8. The seismic qualification is to be in accordance with IEEE 344-1975 methods. A summary of the qualification methods and results of the qualification evaluation is to be provided by the Applicant. Tables 3.10-1 and 3.11-9 have been reviewed in the course of this assessment to determine whether additions to the equipment lists are needed because of the requirements of Regulatory Guide 1.97. Where changes are needed, a statement to that effect is included in the individual paragraphs that follow. Qualification of these additional instruments is the responsibility of the Applicant.

Where performance requirements are required to be specified (implementation position paragraphs C.1.b and C.2.4), the accuracy for various components of each instrument channel have been assessed as they apply to normal operation. Assessment of any potential instrument accuracy variation due to post-accident environments and their relationship to the accuracy needs of the instruments during post-accident periods has not been addressed as part of this appendix. Such an evaluation should be made as part of the environmental qualification program for the post-accident instrumentation and is the Applicant's responsibility to address. As a minimum, Type A variables should be evaluated in this regard.

1D.2.3.1 Neutron Flux

The current neutron monitoring system consists of the source range monitor (SRM), intermediate range monitor (IRM), and the power range monitor (LPRM/APRM). While significant redundancy exists in the design, the current design does not meet the requirement from Subsection C.1.3.1.b to be "... electrically independent and physically separated from each other and from equipment not classified important-to-safety in accordance with Regulatory Guide 1.75." In addition, because the SRM and IRM must be inserted into the core following a reactor scram and because the drive equipment is not likely to survive a post-accident environment for seismic disturbance, the SRM and IRM do not meet the requirements for availability to monitor the variable during the time of interest.

General Electric is currently evaluating the design of a system which will comply with the criteria specified in Regulatory Guide 1.97. The system, called the Wide Range Neutron Monitoring (WRNM) System, will be a replacement for the SRM and IRM indications. Subsequent review and acceptance by the NRC will be required after the system design is finalized.

1D.2.3.2 Control Rod Position Indication

Control Rod Position Indication is classified as Category 3 by the Regulatory Guide because it serves as a backup to neutron flux. Its post-accident monitoring function is only to verify function of a reactor protection system and, consequently, this function is only required for a brief period of time. Because of these reasons, the equipment function time and local environment are not specified for the control rod position indication as called for by the implementation position. No recommendations for change are made for Control Rod Position Indication.

1D.2.3.3 Reactor Coolant Boron Concentration

Boron concentration in the reactor water is determined by analysis of a reactor water sample obtained from the post-accident sample station (refer to Subsection 1D.2.3.38). Recommended boron analysis procedures are included with the operation and maintenance manual supplied with the sample station. Actual procedures are left for the applicant to provide.

1D.2.3.4 RPV Water Level Indication

The existing post-accident water level indication system consists of three wide-range instruments (calibrated for full pressure) covering the range -160" to +60" and two fuel zone instruments (calibrated for atmospheric pressure) covering the range -326" to -116". Narrow-range indication is not included in the assessment.

The RPV water level indication system is the primary variable indicating the availability of adequate core cooling and is considered acceptable, without diverse methods of indication, provided adequate redundancy and unambiguity are provided over the entire range of interest.

The range for coolant level in the reactor specified on Table 1D-1 of the implementation position has been modified to require indication only "above normal reactor water level" rather than the "centerline of main steam line" as specified in the regulatory guide. This change was made because the high reactor water level transient has been shown to be of not great concern (Subsection 1A.23) especially considering the existence of improved high reactor water level trips on HPCS. Furthermore, since the presence of reactor water on the normal range assures adequate core cooling, higher ranges of indication are not needed for this purpose.

1D.2.3.4 RPV Water Level Indication (Continued)

The requirement in Table 1D-1 of the regulatory guide for BWR core thermocouples has been replaced with a requirement that the reactor water level indication should extend from below the core plate to above the pressure vessel Level 1 as adequate indication of core cooling. Through work by General Electric and BWR Owners' Group (References 2 and 3), the NRC staff has recently retracted the requirement for BWR core thermocouples (Reference 4), but they have requested that an investigation continue to identify an acceptable diverse means of indicating adequate core cooling. This investigation is still in progress. Work by General Electric and the BWR Owners' Group (Reference 3) has provided the NRC staff with extensive information on the relationship between the reactor water level and adequate cooling of the core. These studies have shown that as long as at least one of the water delivery systems is available and flowing to the reactor pressure vessel, that adequate core cooling exists. Indication of these flows is already required by the Regulatory Guide. Consequently, it is General Electric's view that no instrumentation other than RPV level indication is required to assure indication of adequate core cooling.

Subsection C.1.3.1.b states that additional instrumentation should be provided to allow the operator to determine the actual conditions of the plant "when failure of one accident monitoring channel results in information ambiguity." Three independent channels of indication are considered acceptable to meet this provision.

During a small-break accident which results in increasing drywell temperatures, the accuracy of the RPV water level indication varies significantly. This effect is described in detail in Reference 2.

A small drywell break could lead to ambiguity in all instrument ranges either because the redundant channels would not agree (if one failed) or because of increasing drywell temperatures and its

1D.2.3.4 RPV Water Level Indication (Continued)

effect on vessel level indication. In addition there could be ambiguity in the overlap region between fuel zone and wide-range indications due to differing calibration conditions and the drywell temperature effects.

These drywell temperature effects are minimized by specifying a minimum vertical elevation drop in the drywell. However, they are not eliminated and some disagreement between channels can be expected.

In addition to not having three independent channels the fuel zone instruments also use nondivisional power as compared with station standby power as required by subsection C.1.3.1.c.

In order to resolve the redundancy and ambiguity problems discussed above, and because of the importance of RPV water level to the verification of adequate core cooling, a three-divisional, unambiguous system of measuring RPV water level is under review. The Enhanced Water Level Indication (ELI) System, which is described in detail in Attachment C, provides three Class 1E channels of water level indication spanning the full range of required reactor water level measurement from the bottom of the core support plate to the top of the current wide indicators. The system uses microprocessors to provide temperature and pressure compensation and to provide a variety of alarms for off normal conditions such as reference leg boiling/flashing or leaking equalizer valves.

1D.2.3.5 RPV Pressure

Two recorded channels of RPV pressure indication are provided in the current design by pressure transmitters connected to the same reference leg as is used for the reactor vessel water level indication. Depending on the location of the break, a small-break

1D.2.3.5 RPV Pressure (Continued)

accident in the instrument line, combined with a power supply failure in the redundant channel, could prevent the operator from having unambiguous indication of reactor pressure. (Nonsafety-related indication of both reactor water level and reactor pressure might be available associated with the feedwater control system indication.) Because RPV pressure is associated with required Manual Safety actions (Type A) unambiguous indication should be made available to meet Subsection C.1.3.1.b of the guide. The ELI design discussed in Attachment C also includes digital indication of a third channel of reactor pressure. This third digital channel has a useful operational benefit for verification of low pressure ECCS injection and for RHR shutdown cooling system initiation permissives.

1D.2.3.6 Drywell Sump Level

The current design consists of two drywell sumps: equipment drain sump for identified leakage, and floor drain sump for unidentified leakage. The monitoring instrumentation is shown in Figures 7.6-12c and 7.6-12d.

An exception is made to the regulatory guide as written for the design category on this variable. Rather than Category 1, General Electric's position is that Category 3 design requirements are more appropriate for the following reasons:

Indication of drywell sump level provides post-accident monitoring of drywell leakage and may be an early indication of a very small drywell break. However, it is primarily a backup variable to other indications of drywell breaks such as drywell pressure (Subsection 1D.2.3.7) or drywell radiation level (Subsection 1D.2.3.8). As such a lower design classification is appropriate.

1D.2.3.6 Drywell Sump Level (Continued)

Both drywell sump discharges are automatically isolated by the primary containment isolation design (Subsection 6.2.4). Thus the only period of meaningful post-accident information is prior to the sump filling to the top. Because of this relatively short duration for any but very small breaks, providing instrumentation to meet the higher design categories is not considered necessary.

In comparison with Category 3 design requirements, the existing design is incomplete in terms of range of indication only for the equipment drain sump. The floor drain sump recorder adequately monitors sump level and rate of change of level from the bottom to the top of the sump as recommended by the guide. A modification is under review which would add sump level indication to the equipment drain sump similar to that used on the floor drain sump. (Figure 7.6-12c).

1D.2.3.7 Drywell Pressure

Two channels of drywell pressure indication are provided in the control room with a range consistent with that identified in the regulatory guide. These instruments are separate from the narrow-range instruments used for the reactor protection system.

These two channels provide adequate redundancy to provide unambiguous indication, even in the event of failure of one of the channels, since diverse indication is provided to indicate a breach of the reactor coolant pressure boundary. These diverse indications include RPV Pressure (Subsection 1D.2.3.5) and Suppression Pool Water Level (Subsection 1D.2.3.9).

No modifications to the 238 Nuclear Island design are necessary. The pressure transmitters and indicators, however, should be added

1D.2.3.7 Drywell Pressure (Continued)

to Tables 3.10-1 (T41) and 3.11-9 (T41) to ensure their qualification.

1D.2.3.8 Containment Area Radiation Monitoring

Area radiation levels at different locations in the containment are defined in Section 12.3. The specific location and r  e of area radiation monitors is the responsibility of the Appli ant to provide.

1D.2.3.9 Suppression Pool Water Level

Indication of water level in the suppression pool is provided by four physically separated (one per pool quadrant) sensors powered from Division 1 and 2 Class 1E power. Additional discussion may be found in Subsections 6.2.7.5, 7.3.1.1.6, and 7.3.2.6.1.

An exception to the range specified in the regulatory guide is taken for this variable. The 238 Nuclear Island design provides indication from about -16'9" elevation to -4'9" elevation. This 12 ft range extends from about 1 foot above the top drywell vent to near the top of the weir wall. For Type C variables (to detect a breach of the coolant pressure boundary) the guide specifies that the range extend from "the bottom of ECCS suction line to 5 feet above normal water level."

This lower level corresponds to about a -27 ft elevation and is about 9 ft below the top drywell vent. Monitoring water level to this low level is considered unnecessary because of the presence in the 238 Nuclear Island design of the upper pool dump feature and the physical arrangement of the plant design which precludes lower water levels.

1D.2.3.9 Suppression Pool Water Level (Continued)

For Type D variables (to monitor operation) the guide specifies "top of vent to top of weir well." The 238 Nuclear Island design is consistent with the range specified to monitor operation of the system.

The upper range is an appropriate upper bound since higher levels would start to overflow into the drywell. Containment flooding, as specified by the Emergency Procedure Guidelines (EPG) (Reference 1, step SP/L-3.5), specifies a maximum primary containment water level which is somewhat higher than the range provided by the current design of level indication. However, in practice, sources of water from external to the primary containment would not be used as called for in EPG step SP/L-3.1. Thus the likelihood of reaching the maximum containment water level is very remote and higher ranges of water level indication are not needed.

1D.2.3.10 Primary Containment Isolation Valve Position

The primary containment isolation valves reviewed in this assessment were identified from a review of Table 6.2-25. All valves providing a containment isolation function (except check valves) were included.

A detailed review of each valve to ensure that valve position indication is provided was not conducted because there are other requirements related to the NRC's Standard Review Plan 6.2.4 which require the availability of control room indication of primary containment isolation valve position. Valve position indication is, in general, provided by limit switches attached to the valve operators which provide a signal to open and close indicating lights on the control room panel.

Containment isolation valve limit switches should be included or referenced by Tables 3.10-1 and 3.11-9 to ensure qualification of the switches.

1D.2.3.10 Primary Containment Isolation Valve Position (Continued)

Implementation position Section C.1.c which is taken from ANSI 4.5 deals with the duration time for equipment qualification. This paragraph states that "shorter qualification times are acceptable if equipment replacement or repair can be accomplished within an acceptable out-of-service time." The primary containment isolation valve position indicating lights in the control room are easily replaceable in a short period of time and a redundant (open or close) indication is provided. Because of these considerations, the valve position indicating lights are viewed as not requiring any specific qualification to meet the intent of the regulatory guide.

1D.2.3.11 Primary Containment Temperature

This variable is not specified by the regulatory guide. However, it has been included as a Type A variable because initiation of the containment spray system is specified by the Emergency Procedure Guidelines (Reference 1), step CN/T) in response to a high indication of this variable. The 238 Nuclear Island design provides two channels of indication on the control room BOP benchboard. The design is shown in Figure 9.4-6.

No modifications are needed to this design. However, the instruments including displays should be included in Tables 3.10-1 and 3.11-9 to ensure qualification.

1D.2.3.12 Primary Containment Pressure

Containment pressure is indicated by two channels of indication as shown in Figure 9.4-6.

The instrumentation provides redundant indication up to twice the design pressure of the containment building. This range is considered adequate to accomplish the functions specified in the Emergency Procedure Guidelines especially considering that a diverse indication of containment temperature is provided.

1D.2.3.12 Primary Containment Pressure (Continued)

Two channels of pressure instrumentation are adequate in case of a single failure because diverse indications to monitor containment integrity are provided.

No modifications are needed for this instrumentation. However, the instruments including displays should be included in Tables 3.10-1 and 3.11-9 to ensure qualification.

1D.2.3.13 Drywell/Containment Hydrogen Concentration

Drywell and containment hydrogen concentration may be determined by analysis of samples obtained from the post-accident sample station (Subsection 1D.2.3.38). On-line instrumentation is the responsibility of the Applicant to provide.

1D.2.3.14 Secondary Containment Area Radiation

Area radiation levels in the secondary containment are defined in Section 12.3. The specific location and range of area radiation monitors are the responsibility of the Applicant to provide.

1D.2.3.15 Secondary Containment Noble Gas Effluent

Applicant to provide.

1D.2.3.16 Containment Noble Gas Effluent

Applicant to provide.

1D.2.3.17 Suppression Pool Temperature

Two temperature sensors per quadrant of the suppression pool are provided in the 238 Nuclear Island design with control room

1D.2.3.17 Suppression Pool Temperature (Continued)

indication and recording. The instruments are discussed in Subsections 7.6.1.11 and 7.6.1.12.

The sensors are located in the upper third of the suppression pool and thus provide a conservative indication of suppression pool temperature for use on the Emergency Procedure Guidelines.

No modifications are needed to these instruments. The instruments including the displays should be included in Tables 3.10-1 and 3.11-9 to ensure qualification.

1D.2.3.18 Drywell Air Temperature

Two Class 1E channels of drywell temperature indication are provided on the control room BOP benchboard (P870) as shown in Figure 9.4-5.

The range of the display (up to 400°F) is adequate to carry out the functions prescribed by the Emergency Procedure Guidelines (Reference 1).

No modifications to the design are needed. The instruments including displays should be included on Tables 3.10-1 and 3.11-9 to ensure qualification.

1D.2.3.19 Coolant Radiation

No instrumentation is provided in the current design to monitor radioactivity levels in the primary coolant and no changes to the plant design are planned.

The specified range for the potential instrument (1/2 Technical Specification Limits (TSL) to 100 times TSL) suggests that the

1D.2.3.19 Coolant Radiation (Continued)

purpose of this instrument is to assess coolant radiation level during routine plant operation. The current design provides sampling capability for reactor coolant as described in Subsection 9.3.2 and provides offgas and mainstream line process radiation measurement as discussed in Section 11.5 for detection of fuel cladding branches.

The value for the technical specification limit has not been established by the staff in standard technical specifications for BWR/6. Subsection 16.3/4.4.5, however, indicates that the TSL is 2 μCi/g Iodine-131 equivalent. On-line reactor coolant monitoring of this level of coolant activity may be impractical during normal operation because of the additional contributions to the detector from other isotopes such as circulating Nitrogen-16 or Cobalt-60 deposited on reactor coolant piping.

Furthermore, should a significant breach of the fuel cladding occur the expected levels of iodine in the reactor coolant would far exceed the upper range specified by the regulatory guide for this instrument. The samples provided by the post-accident sample system (Subsection 1D.2.3.38) will provide quantification of the coolant radioactivity level. Under these conditions, an on-line monitor would serve no useful purpose.

1D.2.3.20 Coolant Gamma Sample

The radioactivity content of the reactor water is determined by analysis of a reactor water sample obtained from the post-accident sample station (Subsection 1D.2.3.38). Recommended procedures to determine the gross activity in the coolant are included with the operation and maintenance manual supplied with the sample station. Actual procedures are the responsibility of the Applicant.

1D.2.3.21 MSIV Leakage Control System Pressure

The current design uses a Class 1E positive leakage control system as described in Section 6.7. Proper system function is monitored and recorded by air system flows rather than system pressure as specified by the regulatory guide. System isolation automatically occurs on high flow or low differential pressure between the RPV and the pressurized lines.

The flow monitoring instrumentation is considered adequate to meet the intent of the regulatory guide to indicate proper system function. No changes are planned for this system.

1D.2.3.22 RHR System Flows

The RHR System serves a variety of functions among them being low pressure coolant injection, containment spray, suppression pool cooling, and shutdown cooling. The valving arrangements (refer to Figure 5.4-12a) required to achieve these different functions of the RHR System occur downstream of the flow element and flow transmitter which is used to indicate RHR System flow. This instrument channel therefore provides the operator with flow indication during any of these operating modes for the RHR System.

From an operational point of view, proper functioning of the containment spray mode of the RHR System is provided by the containment temperature (Subsection 1D.2.3.11) or containment pressure instrumentation (Subsection 1D.2.3.12). Should the containment spray mode be used, it is anticipated that the operators would only initiate flow long enough to decrease these containment parameters at which time flow in the containment spray mode would be terminated. Thus, the primary indicator of proper containment spray mode operation is the containment pressure or temperature indication rather than the RHR System flow.

1D.2.3.22 RHR System Flows (Continued)

Indication of proper LPCI operation is provided by a combination of RHR System flow and valve position indication associated with the injection lines. In addition to the RHR System flow and valve position, an increasing trend on reactor water level indication (Subsection 1D.2.3.4) provides an operator with knowledge of proper LPCI mode function.

No change to RHR System flow indication is planned as a result of this assessment.

1D.2.3.23 RHR Heat Exchanger Outlet Temperature/RHR Service Water Flow

As shown in Figure 5.4-12a the RHR heat exchanger outlet temperature is monitored by temperature elements on the RHR System side and on the service water side of the heat exchanger.

Display of these temperatures is on a commercial quality "Westronics" recorder located in the control room. This instrument does not meet the qualfication standards indicated for Category 2 instruments and and is not included on Table 3.11-9-E12 as needing qualification.

As an alternate to RHR heat exchanger outlet temperature, the RHR service water flow is monitored and displayed in the control room, as shown on Figure 5.4-12a. Monitoring of this variable is an acceptable alternate to outlet temperature since it is an unambiguous indication of heat removal from the RHR primary side especially considering that heat exchanger fouling is provided for in the design. These instruments are included in the List of Specified Instruments (Table 3.11-9-E12).

Monitoring RHR service water flow also satisfies the need for ESF flow monitoring (Subsection 1D.2.3.30). The RHR service water flow

1D.2.3.23 RHR Heat Exchanger Outlet Temperature/RHR Service Water Flow (Continued)

indication has been assumed to satisfy part of that requirement for purposes of this study.

1D.2.3.24 RCIC/HPCS/LPCS Flow

Each of these systems is indicated on the control room by a single channel flow indication system meeting the range specified. No changes are planned for these instruments. The LPCS flow indicator (E21-R600) should be added to Table 3.11.9 (E21) to ensure its qualification.

1D.2.3.25 Standby Liquid Control (SLC) Flow/Pressure

No flow indication is provided in the 238 Nuclear Island design. The positive displacement SLC pumps are designed for constant flow as described in Section 9.3. This flow is periodically tested to ensure it is at the rated value. Any flow blockage or line break would be indicated by abnormal system pressure (high or low) following SLC initiation. The 238 Nuclear Island design includes a safety-related channel of SLC pressure. Changing neutron flux (Subsection 1D.2.3.1), SLC pressure, and squib value position are considered adequate to verify proper system function. Because of these reasons no flow indication is needed and no modifications are planned. SLC pressure and squib valve position are thus considered adequate to meet the regulatory guide as defined in Section C.2.2 of the implementation position. The SLC control room pressure indicator should be included on Tables 3.10-1 and 3.11-9 to ensure qualification.

1D.2.3.26 SLC Tank Level

The 238 Nuclear Island design consists of commercial quality air-powered "dip tube" instrumentation which monitors SLC tank level over the range specified by the regulatory guide. The instrumentation is shown in Figure 9.3-5.

1D.2.3.26 SLC Tank Level (Continued)

An exception is taken to the design category specified by the regulatory guide. Category 3 requirements are considered adequate for this variable since it serves a backup function to other parameters used to monitor system operation (Subsection 1D.2.3.25).

No modifications are needed for the SLC tank level indication channel to meet Category 3 design requirements.

The air supply to the SLC level indication is provided from the instrument air distribution system shown in Figure 9.3-2. The isolation requirements of this air line are currently under review in connection with the design to accommodate anticipated transients without scrams to ensure its availability.

1D.2.3.27 SRV Position Indication

Postive indication of SRV open and closed position is provided by the SRV open/closed monitoring system and is discussed in Section 1A.24.

No modifications to the system as described are planned. The pressure switches and computer logic should be added to Table 3.11-9 (B21) for proper qualification.

1D.2.3.28 Feedwater Flow

Feedwater flow indication is provided by recording the current design as shown in Figures 5.1-3d and 7.7-6.

No modification to the instruments as described are needed. However, since this variable does not indicate the "operation of a safety system or other systems important to safety" its classification as a Type D variable is not justified.

1D.2.3.29 Condensate Storage Tank Level

Applicant to provide.

1D.2.3.30 ESF Cooling Water Flow/Temperature

Cooling water flows and temperatures at all ESF components is indicated by local instruments for the purpose of achieving system balancing (Section 9.2). Qualified control room indication of total system flow and temperature should be provided by the applicant. For compliance with this part of the regulatory guide these instruments are outside of the 238 Nuclear Island scope of supply. The RHR heat exchanger flows and temperatures are discussed in Subsection 1D.2.3.21.

1D.2.3.31 Radwaste High Radioactivity Tank Level

The current Radwaste System design is described in Section 11.2. Level indication of three high conductivity tanks and two low conductivity tanks is provided by recorders in the radwaste control room.

No modifications are necessary to this instrumentation to meet the Regulatory Guide 1.97 criteria. However, since the Radwaste System is neither a safety system nor a system important to safety, its classification as Type D is not justified.

1D.2.3.32 Emergency Vent Damper Position

Indicating lights showing the vent damper position on all supply and exhaust paths used during normal operation for the Containment, Auxiliary Fuel buildings are provided on the control room BOP benchboard. Radwaste building ventilation supply and exhaust damper position is indicated in the radwaste control room.

No modifications are needed to comply with the criteria of the regulatory guide. The position indicator switches should be included on Tables 3.10-1 and 3.11-9 to ensure qualification.

1D.2.3.33 Standby Power Sources

The 238 Nuclear Island design includes control room indication of electrical, air, and liquid power supply systems.

The electrical power supply instrumentation is shown on Figures 8.3-2, 8.3-3, and 8.3-15. Safety-related control room display (voltage and amperes) of three 6.9 kV ESF buses and four 480V Class 1E buses supplied from 6.9 kV feeders (voltage only) is provided. In addition, the design includes voltage display of four Class 1E 125V direct current buses. Control room indication of the electrical operation (amperes, voltage, watts, vars, and frequency) of each of the three diesel generators is also provided. No changes are needed to provide indication of the electrical system status during post-accident periods.

Air supplies to safety-related valves and functions are provided by the Pneumatic Supply System discussion in Section 6.8. Non-safety-related valves and instruments are powered by the compressed air systems described in Subsection 9.3.1.

Only the safety-related portion of the Pneumatic Supply System needs to be monitored under post-accident conditions. The safety-related functions are identified on Table 6.8-1. Two channels of control room indication are provided on the 238 Nuclear Island design. The pressure transmitters and indications should be included on Tables 3.10-1 and 3.11-9 to ensure qualification.

The air supply to the Air Positive Seal Isolation Valve Leakage Control System and the water supply to the Water Positive Seal Isolation Valve Leakage Control System are described in Subsection 6.5.3.3.

The Positive Seal Leakage Control Systems provides a backup function to the Primary Containment Isolation System which is monitored by the individual isolation valve position indications (Subsection 1D.2.3.10). For this reason Category 3 design requirements are applied to these channels.

1D.2.3.34 Ventilation Flow Rate

Post-accident ventilation from the auxiliary, fuel, and shield building air spaces is through the Standby Gas Treatment System (SGTS). All other exhaust flows (except for those from the Radwaste Building) are isolated. Consequently only SGTS flow needs monitoring.

No instrument to specifically monitor SGTS flow is provided in the 238 Nuclear Island design except for isometric probes on the SGTS vent which are the responsibility of the applicant to provide.

The design does provide safety grade indication of SGTS damper position on the control room BOP benchboard: Radwaste ventilation damper position is provided in the Radwaste Control Room. These position indications are considered adequate to meet the intent of the regulatory guide.

1D.2.3.35 Particulate/Halogen Release

Applicant to provide.

1D.2.3.36 Environs Radioactivity Monitoring

Applicant to provide.

1D.2.3.37 Meteorology

Applicant to provide.

1D.2.3.38 Post-Accident Sampling

Post-accident samples are obtained from the post-accident sample station (PASS) designed to meet NUREG 0737 requirements. The PASS is described in Subsection 1A.21.

1D.2.3.38 Post-Accident Sampling (Continued)

An exception is taken to the areas requiring sampling specified by the regulatory guide. Samples of drywell, containment, auxiliary or fuel building sumps are not provided for in the current design and no modifications are planned. This position is justified because no useful information such as the extent of core damage, or for release assessment would be obtained through such samples.

Flow to these sumps is primarily from reactor water leakage either planned or unplanned from breaks. The PASS samples reactor water directly thus provides an indication of the extent of possible core damage. Leaks of radioactive liquid from the systems which could cause increases in sump level are primarily detected by area and process radiation monitors (Subsections 1D.2.3.8, 1D.2.3.14 and 1D.2.3.15). These indications are a direct indication of potential gaseous offsite impact.

These sumps are automatically isolated by a containment isolation signal (Subsection 6.2.4). Since any potential release is contained by this action, liquid release outside the contained areas is not likely. Because of these reasons samples of the sumps are not necessary.

In addition to being unnecessary, sump samples may be impractical because of the small sample volume (0.1 ml) and the high particulate content expected in any sump. Without extensive filtering, clogging of the liquid sample ball valve is a possibility.

## 1D.3 REFERENCES

1. Emergency Procedure Guidelines - BWR 1-6, January 1981 (NEDO-24934).

2. Review of BWR Reactor Vessel Water Measurement, April 1980 (NEDE-24801) (Proprietary).

3. Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors Volume 1, August 1979 (NEDO-24708A).

4. NRC Staff Recommendations on the Requirements for Emergency Response Capability, March 10, 1982.

Table 1D-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF 238 NUCLEAR ISLAND

| Variable | Type* | Category* | Discussion Subsection |
|---|---|---|---|
| Reactivity Control | | | |
| Neutron Flux (value, rate, trend) | A,B | 1 | 1D.2.3.1 |
| Control Rod Position | B | 3 | 1D.2.3.2 |
| Boron Concentration (sample) | B | 3 | 1D.2.3.3 |
| Core Cooling | | | |
| Coolant Level in the Reactor (value, trend) | A,B,C | 1 | 1D.2.3.4 |
| Maintaining Reactor Coolant System Integrity | | | |
| RCS Pressure (value + alarm) | A,B,C | 1 | 1D.2.3.5 |
| Drywell Sump Level (value + alarm) | B,C | 3 | 1D.2.3.6 |
| Drywell Pressure (value + alarm) | B,C,D | 1,2 | 1D.2.3.7 |
| Primary Containment Area Radiation | E<br>C | 1<br>3 | 1D.2.3.8 |
| Suppression Pool Water Level | A,C,D | 1,2 | 1D.2.3.9 |
| Maintaining Containment Integrity | | | |
| Primary Containment Isolation Valve Position (Excluding Check Valves) | B | 1 | 1D.2.3.10 |
| Primary Containment Temperature | A | 1 | 1D.2.3.11 |

*As defined by Attachment A

Table 1D-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF 238 NUCLEAR ISLAND (Continued)

| Variable | Type* | Category* | Discussion Subsection |
|---|---|---|---|
| Maintaining Containment Integrity (Continued) | | | |
| Primary Containment Pressure (value, rate, trend, + alarm) | A,B,C | 1 | 1D.2.3.12 |
| Drywell/Containment Hydrogen Concentration (value) | A,C | 1 | 1D.2.3.13 |
| Secondary Containment Area Radiation (value) | C,E | 2 | 1D.2.3.14 |
| Secondary Containment Noble Gas Effluent | C,E | 2 | 1D.2.3.15 |
| Primary Containment Noble Gas Effluent | C | 3 | 1D.2.3.16 |
| Suppression Pool Temperature | A,D | 1,2 | 1D.2.3.17 |
| Drywell Air Temperature | A,D | 1,2 | 1D.2.3.18 |
| Fuel Cladding Barrier Monitoring | | | |
| Coolant Radiation (value + alarm) | NA | NA | 1D.2.3.19 |
| Coolant Gamma (1 sample/4 hours) results within 72 hr | C | 3 | 1D.2.3.20 |
| System Operation | | | |
| Main Steam Line Isolation Valve Leakage Control System Pressure | D | 2 | 1D.2.3.21 |
| Containment Spray Flow | D | 2 | 1D.2.3.22 |

*As defined by Attachment A

Table 1D-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF 238 NUCLEAR ISLAND (Continued)

| Variable | Type* | Category* | Discussion Subsection |
|---|---|---|---|
| System Operation (Continued) | | | |
| Residual Heat Removal (RHR) System Flow | D | 2 | 1D.2.3.22 |
| RHR Service Water Flow | D | 2 | 1D.2.3.23 |
| Low Pressure Coolant Injection System Flow | D | 2 | 1D.2.3.22 |
| Reactor Core Isolation Cooling System Flow | D | 2 | 1D.2.3.24 |
| High Pressure Coolant Spray System Flow | D | 2 | 1D.2.3.24 |
| Core Spray System Flow | D | 2 | 1D.2.3.24 |
| Standby Liquid Control System (SLCS) Flow | D | 2 | 1D.2.3.25 |
| SLCS Storage Tank Level | D | 3 | 1D.2.3.26 |
| SRV Position | D | 2 | 1D.2.3.27 |
| Feedwater Flow | D | 3 | 1D.2.3.28 |
| CST Level | D | 3 | 1D.2.3.29 |
| ESF Cooling Water Flow | D | 2 | 1D.2.3.30 |
| ESF Cooling Water Temperature | D | 2 | 1D.2.3.30 |
| High Radioactivity Tank Level | D | 3 | 1D.2.3.31 |
| Emergency Vent Damper Position | D | 2 | 1D.2.3.32 |
| Standby Energy Status | D | 2 | 1D.2.3.33 |

*As defined by Attachment A

Table 1D-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT
OF 238 NUCLEAR ISLAND (Continued)

| Variable | Type* | Category* | Discussion Subsection |
|---|---|---|---|
| Effluent Monitoring | | | |
| SGTS Ventilation Flow Rate | E | 2 | 1D.2.3.34 |
| Other Ventilation Flow Rates | E | 3 | 1D.2.3.34 |
| Particulate/Halogen Release (sample) | E | 3 | 1D.2.3.35 |
| Environs Radioactivity Monitoring | E | 3 | 1D.2.3.36 |
| Meteorology | E | 3 | 1D.2.3.37 |
| Post-Accident Sampling (sample) | E | 3 | 1D.2.3.38 |

*As defined by Attachment A

Table 1D-2

SUMMARY OF INFORMATION INDICATING DEGREE OF COMPLIANCE OF 238 NUCLEAR ISLAND WITH REGULATORY GUIDE 1.97, REVISION 2

| Variable | Type | Qualification | | Quality Assurance | Redundancy | Range | Power Supply | Control Room Display | Comments/ Notes |
|---|---|---|---|---|---|---|---|---|---|
| | | Environment | Seismic | | | | | | |
| Reactivity Control | | | | | | | | | |
| Neutron Flux | A,B | | | | | | | | |
| Power Range | | RG 1.89 | RG 1.100 | Yes | 4 channels | $10^{12}$nv to $10^{14}$nv | 1E Unint DC | Recorder | A |
| Int Range | | RG 1.89 | RG 1.100 | Yes | 4 channels | $10^{8}$nv to $10^{13}$nv | RPS | Recorder | A,B |
| Source Range | | RG 1.89 | RG 1.100 | No | 4 channels | $10^{3}$nv to $10^{9}$nv | RPS | Recorder | A,B |
| Control Rod Position | B | N/A | N/A | N/A | 1 channel | Full in/full out | Uninterruptible | Full core | |
| Boron Concentration | B | N/A | N/A | N/A | N/A | 0-1000 ppm | N/A | N/A | C |
| Core Cooling | | | | | | | | | |
| RPV Water Level | A,B,C | | | | | | | | |
| Wide Range | | RG 1.89 | RG 1.100 | Yes | 3 channels | -160" to +60" | 1E | Recorders/Indic | D |
| Fuel Zone | | RG 1.89 | RG 1.100 | No | 2 channels | -326" to -116" | Inst Bus | Recorders/Indic | D,E |
| In-Core Thermocouples | | N/A | N/A | N/A | N/A | N/A | N/A | N/A | D |
| Reactor Coolant Integrity | | | | | | | | | |
| RPV Pressure | A,B,C | RG 1.89 | RG 1.100 | Yes | 2 channels | 0-1500 psig | 1E | Recorders | E,J |
| DW Equip Dr Sump Level | B,C | N/A | N/A | N/A | 1 channel | High-high | Inst Bus | Alarm | F,L |
| DW Flow Dr Sump Flow | B,C | N/A | N/A | N/A | 1 channel | 0-5 gpm | Inst Bus | Recorder | L |
| Drywell Pressure | B,C,D | RG 1.89 | RG 1.100 | Yes | 2 channels | 0 to 30 psig | 1E | Indicator | A,J |
| Pri Cont Radiation | C,E | * | * | * | * | * | * | * | |
| Supp Pool Water Level | A,C,D | RG 1.89 | RG 1.100 | Yes | 1 channel per each of 4 quadrants | -16'9" to -4'9" | 1E | Recorder/Indic | A |
| Containment Integrity | | | | | | | | | |
| Isolation Valve Position | B | RG 1.89 | RG 1.100 | Yes | 2 channels per each of 2 values | Open/closed | 1E | Indicator lights | A |
| Cont Temperature | A | RG 1.89 | RG 1.100 | Yes | 2 channels | 0 to 300°F | 1E | Indicator | A |
| Cont Pressure | A,B,C | RG 1.89 | RG 1.100 | Yes | 2 channels | 0 to 30 psig | 1E | Indicator | A,H,J |
| DW/Cont Hydrogen | A,C | * | * | * | * | * | * | * | |
| Sec Cont Area Rad | C,E | N/A | N/A | N/A | * | * | * | * | |
| Sec Cont Effluent | C,E | * | * | * | * | * | * | * | |
| Pri Cont Effluent | C | N/A | N/A | N/A | * | * | * | * | |
| Supp Pool Temp | A,D | RG 1.89 | RG 1.100 | Yes | 2 channels per each of 4 quadrants | 0 to 300°F | 1E | Indicator/ Recorder | A |
| DW Temperature | A,D | RG 1.89 | RG 1.100 | Yes | 2 channels | 0 to 400°F | 1E | Indicator | A,H |

*Applicant to provide.

## Table 1D-2

## SUMMARY OF INFORMATION INDICATING DEGREE OF COMPLIANCE OF 238 NUCLEAR ISLAND WITH REGULATORY GUIDE 1.97, REVISION 2 (Continued)

| Variable | Type | Qualification Environment | Qualification Seismic | Quality Assurance | Redundancy | Range | Power Supply | Control Room Display | Comments/Notes |
|---|---|---|---|---|---|---|---|---|---|
| Fuel Cladding Barrier | | | | | | | | | |
| Coolant Radiation | N/A | N/A | | N/A | N/A | N/A | N/A | N/A | C |
| Coolant Gamma (sample) | C | N/A | | N/A | N/A | N/A | N/A | N/A | C |
| System Operation | | | | | | | | | |
| MSIV LCS Flow | D | RG 1.89 | RG 1.100 | Yes | Inboard/ Outboard System per Each Steam Line + Drain | 0 to 15 scfm | 1E | Recorder | |
| Cont Spray Flow | D | N/A | | N/A | N/A | N/A | N/A | N/A | See RHR Flow |
| RHR Flow | D | RG 1.89 | RG 1.100 | Yes | 1 channel per each of 3 loops | 0 to 10,000 gpm | 1E | Indicator | |
| RHR Service Water Flow | D | RG 1.89 | RG 1.100 | Yes | 1 channel per each of 2 loops | 0 to 10,000 gpm | 1E | Indicator | G |
| LPCI Flow | D | N/A | | N/A | N/A | N/A | N/A | N/A | See RHR Flow |
| RCIC Flow | D | RG 1.89 | RG 1.100 | Yes | 1 channel | 0 to 800 gpm | 1E | Indicator | |
| HPCS Flow | D | RG 1.89 | RG 1.100 | Yes | 1 channel | 0 to 10,000 gpm | 1E | Indicator | |
| LPCS Flow | D | RG 1.89 | RG 1.100 | Yes | 1 channel | 0 to 10,000 gpm | 1E | Indicator | |
| SLC Pressure | D | RG 1.89 | RG 1.100 | Yes | 1 channel | 0 to 1800 psig | 1E | Indicator | A,M |
| SLC Tank Level | D | N/A | N/A | N/A | 1 channel | 0 to 5000 gal | Inst Bus | Indicator | L |
| SRV Position | D | RG 1.89 | RG 1.100 | Yes | 3 channels per SRV | Open/Closed | 1E | Indicator Lights | A |
| Feedwater Flow | D | N/A | N/A | N/A | 1 channel per each of 2 loops | 0 to $20 \times 10^6$ lb/hr | Instrument Bus | Recorder | |
| Cond Storage Tank Level | D | * | * | * | * | * | * | * | |
| ESF Cooling Water Flow | D | * | * | * | * | * | * | * | |
| ESF Cooling Water Temperature | D | * | * | * | * | * | * | * | |
| High Rad Tank Level | D | N/A | N/A | N/A | 1 channel per each of 5 tanks | Bottom to top | Inst Bus | Recorders | I |
| Emerg Vent Damper Position | D | RG 1.89 | RG 1.100 | Yes | 2 channels per each of 2 dampers per each of 11 paths | Open/Closed | 1E | Indicator Lights | A |

*Applicant to provide.

Table 1D-2

SUMMARY OF INFORMATION INDICATING DEGREE OF COMPLIANCE OF 238 NUCLEAR ISLAND WITH REGULATORY GUIDE 1.97, REVISION 2 (Continued)

| Variable | Type | Qualification | | Quality Assurance | Redundancy | Range | Power Supply | Control Room Display | Comments/ Notes |
|---|---|---|---|---|---|---|---|---|---|
| | | Environment | Seismic | | | | | | |
| System Operation (Continued) | | | | | | | | | |
| Standby Power Status | D | RG 1.89 | RG 1.100 | Yes | 1 channel per bus | Voltage, amperes | 1E | Indicators | A |
| Standby Air Status | D | RG 1.89 | RG 1.100 | | 1 channel per each of 2 divisions | 0 to 300 psig | 1E | Indicator | A |
| Standby LCS APS Pressure | D | N/A | N/A | N/A | 1 channel per each of 2 divisions | 0 to 150 psig | 1E | Indicator | L |
| Standby LCS WPS Pressure | D | N/A | N/A | N/A | 1 channel per each of two divisions | 0 to 150 psig | 1E | Indicator | L |
| Effluent Monitoring | | | | | | | | | |
| SGTS Vent Flow | E | RG 1.89 | RG 1.100 | Yes | 2 channels per each of 2 dampers per each of 2 trains | Open/Closed | 1E | Indicator Lights | K |
| Radwaste Vent Flow | E | N/A | N/A | N/A | 2 channels per damper | Open/Closed | Inst Bus | Indicator Lights | I,K,L |
| Part/Halogen Sample | E | * | * | * | * | * | * | * | |
| Environs Monitoring | E | * | * | * | * | * | * | * | |
| Meteorology | E | * | * | * | * | * | * | * | |
| Post-Accident Sample | E | N/A | N/A | N/A | 2 points per area sampled | * | Inst Bus | N/A | N |

*Applicant to provide.

Table 1D-2

NOTES

A. Qualification extends from sensor to display. Addition of instruments to Table 3.10-1 or Table 3.11-9 required.

B. Sensor drive mechanism not qualified or qualifiable to RG 1.89 or RG 1.100 standards. No modification planned. A wide-range instrument is under development as a replacement for these instruments.

C. Sampling is by the post-accident sample station. Recommended analytical procedures provided.

D. An enhanced water level indication is under review as a replacement for these instruments.

E. An additional channel of display is under review.

F. Modification of equipment drain sump instrumentation is under review.

G. This variable is an alternate to RHR heat exchanger outlet temperature.

H. This instrumentation provides adequate information for the plant operator to carry out actions defined by the Emergency Procedure Guidelines.

I. Display is in the Radwaste Building Control Room.

J. Two channels of pressure instrumentation are considered adequate to provide unambiguous post-accident indication.

K. Indication of damper position is sufficient indication of system flow.

L. Design to Category 3 requirements is acceptable for this instrument since it serves a backup function to other variables.

M. SLC pressure provides adequate indication of system operation in combination with other variables.

N. Analytical procedures and laboratory facilities to be provided by Applicant.

Table 1D-3

238 NUCLEAR ISLAND TYPE A VARIABLES

Neutron Flux

RPV Water Level

RPV Pressure

Suppression Pool Temperature

Suppression Pool Water Level

Drywell/Containment Hydrogen Concentration

Drywell Air Temperature

Containment Pressure

Containment Air Temperature

ATTACHMENTS TO
APPENDIX 1D

# ATTACHMENT A TO APPENDIX 1D

# IMPLEMENTATION POSITION

# FOR

# REGULATORY GUIDE 1.97, REVISION 2

# ATTACHMENT A
# TO
# APPENDIX 1D

## IMPLEMENTATION POSITION FOR REGULATORY GUIDE 1.97, REVISION 2

This attachment addresses the criteria for assessment of variables against the requirements* of Regulatory Guide 1.97, Revision 2. The organization of this implementation position follows Regulatory Guide 1.97, Revision 2, Section C, "Regulatory Position," point by point. Some rewording has been made for clarity.

Where other documents are referenced by paragraph in the regulatory guide, the referenced material is included.

Additional clarifying information or modifications to Regulatory Guide 1.97 to reflect an exception to the regulatory guide text are included in brackets [ ].

This attachment constitutes an engineering and licensing position for assessment of 238 Nuclear Island variables.

Table 1DA-1 lists the variables addressed and requirements unique to that variable. Table 1DA-2 summarizes the generic requirements applicable to each category of variable. It should be noted that some of the requirements are Applicant's scope, as indicated in Table 1DA-2 and are to be addressed separately.

*Although the term "requirement" is used throughout this document for clarity, it is recognized that Regulatory Guide 1.97 provides guidance and not requirements.

Position C.1 - Accident Monitoring Instrumentation

[The following criteria, in addition to Positions C.1.1 through C.2.5, should be implemented]:

C.1.a - Regulatory Guide Definition - Type B Variables

Type B Variable: Those variables that provide information to indicate whether plant safety functions are being accomplished. Plant safety functions are (1) reactivity control, (2) core cooling, (3) maintaining reactor coolant system integrity, and (4) maintaining containment integrity (including radioactive effluent control). Variables are listed with designated ranges and category for design and qualification requirements. Key variables are indicated by design and qualification Category 1.

Interpretation/Implementation Position

For the BWR, the Type B variables identified in Table 1DA-1 satisfy the provisions of Regulatory Guide 1.97 and should be implemented as identified in Table 1DA-1 of this implementation position with the exceptions noted. Exceptions are bracketed, [ ].

C.1.b - Performance Requirements

The determination of the performance requirements for Type A, B, and C variable measuring instruments for accident monitoring instrumentation channels should include, as a minimum, identification of:

1. Range of the process variables to be measured and monitoring instrumentation [per Table 1DA-1]

2. Required accuracy of measurement [per Table 1DA-1]

C.1.b - Performance Requirements (Continued)

3. Required response characteristics, if applicable [per Table 1DA-1]

4. Time interval during which the measurement is needed (function time)

5. Local environment(s) in which the instrument channel components must operate

6. Requirements for rate or trend information [per Table 1DA-1]

7. Any required spatial distribution of sensors

8. Any requirements for group displays of related information

C.1.c - Qualification Duration Requirements

The qualification duration of Types A, B, and C, Category 1 and Category 2 information display channels should be:

(1) For Type A variable monitoring instruments - the duration for which the instrumentation is required for manual operator actions to bring the plant to "cold shutdown" following a design basis accident event.

(2) For Types B, C, D, and E variable monitoring instrumentation - at least 100 days following a DBA event unless shorter duration times can be justified and documented.

C.1.c - Qualification Duration Requirements (Continued)

Shorter qualification times are acceptable if equipment replacement or repair can be accomplished within an acceptable out-of-service time, taking into consideration the environment where the equipment is located and the information needs of the operator.

Position C.1.1 - Regulatory Guide Definition

Type A Variables

Those variables that are to be monitored provide the primary information required to permit the control room operator to take the specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accident events. Primary information is information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may be identified in written procedures.

Interpretation/Implementation Position

Type A variables are limited to those variables which are necessary (primary) to alert the control room operator of the need to perform preplanned manual actions for safety systems to perform their safety functions, such as (1) initiating safety systems (e.g., hydrogen mixing, main steam leakage control system) and/or (2) changing safety system lineups, as necessary, to permit the systems to perform safety function (e.g., suppression pool cooling, containment spray, etc.), for which no automatic system controls are provided and are required to mitigate the consequences of specific design basis accidents (DBAs) which threaten the health and safety of the public, as defined in this Safety

Interpretation/Implementation Position (Continued)

Analysis Report (SAR), to bring the plant to a safe condition (e.g., cold shutdown).

Type A variables do not include variables (1) which may indicate whether a specific safety function is being accomplished (Type B) or (2) which may indicate the need for contingency or corrective actions, resulting from the failure of the plant (Type C) or system(s) (Type D) to respond correctly when needed, or (3) which may indicate to the operator that it is desirable to change/modify the operation/alignment of systems important to safety to maintain the plant in a safe condition after plant safety has been achieved. [Emergency actions specified by Emergency Procedure Guidelines (EPGs) in response to specific operating limits should, however, be considered.]

Position C.1.2 - Regulatory Guide Definition

Type C Variables

Those variables that provide information to indicate the potential for being breached or the actual breach of the barriers to fission product releases. The barriers are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.

Interpretation/Implementation Position

Type C variables identified in Table 1DA-1 are considered adequate to satisfy the provisions of Regulatory Guide 1.97 and should be implemented as identified in the Regulatory Guide 1.97 Table 1DA-1 with the exceptions noted in Table 1DA-1 of this implementation position. Exceptions are bracketed, [ ].

Position C.1.3 - Design and Qualification Criteria

C.1.3.1 - Design and Qualification Criteria - Category 1

C.1.3.1.a

Instrumentation should be qualified in accordance with Regulatory Guide 1.89 and the methodology described in NUREG-0588. Qualification applies to the complete instrument channel from sensor to display, where the display is a direct-indicating meter or recording device. Qualification applies to the instrument channel from the sensor to and including the channel isolation device, where the instrumentation channel signal is to be used in a computer-based display, recording, and/or diagnostics program, unless it can be shown that failure of the isolation device cannot jeopardize the function of the Class 1E instrument channel.

The signal isolation device should be located where it is accessible for maintenance during accident conditions.

The portion of the instrumentation channel requiring seismic qualification should be qualified in accordance with Regulatory Guide 1.100 (including hydrodynamic loads). Instrumentation should continue to read within the required accuracy following, but not necessarily during, a safe shutdown earthquake (SSE) (including hydrodynamic loads).

Instrumentation whose ranges are required to extend beyond those ranges calculated in the most severe design basis accident event for a given variable should be qualified as follows:

> The qualification environment for Type C information display channel components shall be based on the plant unique design basis accident events, except those components directly subjected to the monitored variable environment

C.1.3.1.a (Continued)

shall be qualified to the assumed maximum range for the monitored variable. The monitored variable shall be assumed to approach this peak by extrapolating the most severe initial ramp associated with the DBA events. The decay for this variable shall be considered proportional to the decay for this variable associated with the DBA events (see Figure 1DA-1). No additional qualification margin needs to be added to the extended range variable. All environmental envelopes, except those pertaining to the variable measured by the information display channel, shall be associated with the DBA events.*

C.1.3.1.b

Sufficient instrumentation channels should be provided to assure that no single failure (1) within the accident-monitoring channel, or (2) its auxiliary supporting features, or (3) its power sources, concurrent with the failures that are a condition or results of a specific accident (consequential damage) will prevent the operators from being presented the information necessary for them to (1) determine the safety status of the plant and (2) bring the plant to and maintain it in a safe condition following an accident.

*The above environmental qualification requirement for Type C equipment does not account for steady state elevated levels that may occur in other environmental parameters associated with the extended range variable. For example, a Type C sensor measuring containment pressure must be qualified for the measured process variable range (i.e., three times design pressure for concrete containments), but the corresponding ambient temperature is not mechanically linked to that pressure. Rather, the ambient temperature value is the bounding value for design basis accident events analyzed in Chapter 15. The extended range requirement is to ensure that the equipment will continue to provide information if conditions degrade beyond those postulated in the safety analysis. Since Type C variable ranges are non-mechanistically determined, extension of associated parameter levels is not justifiable and has therefore not be required.

C.1.3.1.b (Continued)

Additional instrumentation channels should be provided to allow operators to determine the actual condition of the plant when failure of one accident monitoring channel would result in information ambiguity (viz., redundant displays disagree) that could lead operators to defeat or fail to accomplish the required safety function. Additional instrumentation channels should consist of either (1) providing another independent channel monitoring the same variable (redundancy), or (2) providing an independent channel to monitor a different variable that bears a known relationship to the multiple channels (diversity).

Redundant or diverse channels should be electrically and physically separated from each other and from equipment not classified important to safety in accordance with Regulatory Guide 1.75.

At least one instrument channel should be displayed on a direct-indicating or recording device (e.g., strip chart recorder).

Redundant [or diverse] monitoring channels are not needed within each redundant division of a safety system.

C.1.3.1.c

Instrumentation channels should be energized from station standby power sources as provided in Regulatory Guide 1.32 and should be backed up by batteries where momentary interruption is not tolerable (uninterruptible power).

C.1.3.1.d

Instrumentation channels should be available prior to an accident except as provided in Paragraph 4.11, "Exemption," as defined in IEEE Standard 279 or specified in Technical Specifications.

C.1.3.1.e

The recommendations of the following Regulatory Guides pertaining to quality assurance should be followed:

| | |
|---|---|
| Regulatory Guide 1.123 | Regulatory Guide 1.38 |
| Regulatory Guide 1.144 | Regulatory Guide 1.58 |
| Regulatory Guide 1.146 | Regulatory Guide 1.64 |
| Regulatory Guide 1.28 | Regulatory Guide 1.74 |
| Regulatory Guide 1.30 | Regulatory Guide 1.88 |

[Quality assurance per NEDO-11209, NEBG BWR QA Program description is acceptable.]

C.1.3.1.f

Continuous instrumentation channel readout should be provided (this may be by [either an analog or digital meter] or recorder). [Interruption of instrumentation readout is acceptable where interruptible power is acceptable. (See Position C.1.3.1.c.)]

Overlapping instrument spans should be provided when two or more instruments are required to cover a particular range.

C.1.3.1.g

Recording of instrumentation readout information should be provided as follows:

(1) the information should be continuously available on dedicated recorders [e.g., strip chart recorders], where direct and immediate trend or transient information is essential for operator information or action; or

C.1.3.1.g (Continued)

(2) the information should be continuously updated, stored in a computer memory and displayed on demand where the information is not essential for operator information or action; or

(3) intermittent displays such as data loggers and scanning recorders should be used if no significant transient response information is likely to be lost by such devices.

C.1.3.2 - Design and Qualification Criteria - Category 2

C.1.3.2.a

Instrumentation should be qualified in accordance with Regulatory Guide 1.89 and methodology described in NUREG-0588. [Qualification applies to the complete instrument channel from sensor to display where the display is a direct indicating meter or recording devices.] Qualification applies from the sensor [to and including] the isolator/input buffer where the channel signal is to be processed or displayed on demand, unless it can be shown that failure of the isolator/input buffer cannot jeopardize the function of Class 1E instrument channel.

The signal isolation device should be located where it is accessible for maintenance during accident condition.

The portion of the instrumentation channel requiring seismic qualification should be qualified in accordance with Regulatory Guide 1.100 (including hydrodynamic loads) when the instrumentation is part of a safety-related system.

C.1.3.2.b

Instrumentation channels should be energized from a high-reliability power source, not necessarily standby power, and should be backed up by batteries where momentary interruptions are not tolerable (noninterruptible power).

C.1.3.2.c

Instrumentation channels should be available with the out-of-service intervals based on:

(1) the normal Technical Specification requirements on out-of-service for the system it serves where applicable, or

(2) where specified by other requirements.

C.1.3.2.d

The recommendations of the Regulatory Guide pertaining to quality assurance, identified in Position 1.3.1.e, should be followed, where applicable, considering the importance to safety of the instrumentations under consideration. Since some instrumentation is less important to safety than other instrumentation, the quality assurance requirements that are implemented should:

(1) provide control over activities affecting quality to the extent consistent with the importance to safety of the instrumentation, and

(2) be determined and documented by personnel knowledgeable in the use of the instrumentation.

C.1.3.2.e

The instrumentation readout should be on an individual meter/ recorder (e.g., strip chart recorder) or processed for display on demand by a CRT or other appropriate means.

C.1.3.2.f

Effluent radioactivity monitoring, area radiation monitoring, and meteorology monitoring instrumentation readouts should be:

(1) continuously available on dedicated recorders (e.g., strip chart recorders) where direct and immediate trend or transient information is essential for operator information or action, or

(2) continuously updated, stored in a computer memory, and displayed on demand, if not essential to the operator, and

(3) displayed by dial, digital, CRT or strip chart recorders.

C.1.3.3 - Design and Qualification Criteria - Category 3

C.1.3.3.a

Instrumentation should be of high-quality commercial grade and should be selected to withstand the specified service environment.

C.1.3.3.b

The method of display may be dial, digital, CRT or strip chart recorder indication.

C.1.3.3.b (Continued)

Effluent radioactivity monitoring, area radiation monitoring, and meteorology monitoring instrumentation readouts should be:

(1) continuously available on dedicated recorders (e.g., strip chart recorders) where direct and immediate trend or transient information is essential for operator information; or

(2) continuously updated, stored in computer memory, and displayed or demand, if not essential to the operator; and

(3) displayed by dial, digital, CRT, or strip chart recorders.

Position C.1.4 - Additional Criteria - Categories 1 and 2

C.1.4.a

Category 1 and Category 2 equipment should be designated as part of the accident-monitoring instrumentation [Type A, B, C] or systems operation and effluent-monitoring instrumentation [Type D, E].

Transmission of signals from such equipment for other uses should be through isolation devices that are designated as part of the monitoring instrumentation and that meets the provisions of Regulatory Guide 1.97.

C.1.4.b

Category 1 and 2 instruments designated as Types A, B, and C should be [located visually accessible to the operator] and

C.1.4.b (Continued)

specifically identified on control panels so the operator can easily discern that they are intended for use under accident conditions.

Position C.1.5 - Additional Criteria - Categories 1, 2, and 3

1.5.a

Servicing, testing, and calibration programs [consistent with the requirements of Position 1.3] should be specified to maintain the capability of the monitoring instrumentation. The capability for testing during power operation should be provided for those instruments where the required interval between testing is less than the normal time interval between generating station shutdowns.

[Applicant to provide programs.]

1.5.b

The instrumentation design should facilitate administrative control over removing channels from service.

[Applicant to provide administrative controls.]

1.5.c

The instrumentation design should facilitate administrative control of access to setpoint adjustments, module calibration adjustments, and test points.

[Applicant to provide administrative controls.]

1.5.d

Human factors analysis should be used in determining the type and location of displays and the monitoring instrumentation design should minimize the development of conditions that would cause meters, annunciators, recorders, alarms, etc. to give ambiguous indication potentially confusing to the operator.

[See Chapter 18 regarding conduct of human factors review.]

C.1.5.e

Instrumentation should be designed to facilitate the recognition, location, repair, or adjustment of malfunctioning components or modules.

C.1.5.f

Monitoring instrumentation should be from sensors that directly measure the desired variables whenever possible. Indirect measurements should be made only when they can be shown by analysis to provide unambiguous information.

C.1.5.g

The same instruments should be used for accident monitoring as those used for normal plant operations to enable the operators to monitor instruments with which they are most familiar during accident situations. Separate instruments should be used only when the required "accident" range would result in a loss of instrumentation sensitivity in the normal operating range.

C.1.5.h

Instrumentation channel periodic checks, testing and calibration verification should be in accordance with the applicable portions of Regulatory Guide 1.118 (Note: Instrumentation response time testing for post-accident instrumentation is [not required as it is peak values of the variables that are of interest rather than the rate of peak value achievement].

[Applicant to provide programs.]

Position C.1.6 - Additional Criteria - Type B and C Variables

C.1.6.1 - Type B Variables

In conjunction with Table 1DA-1 the following should be considered as a minimum number of instruments, and their respective ranges, for accident monitoring instrumentation.

C.1.6.1.a - Reactivity Control Monitoring

The measured variable should be neutron flux or combinations of other variables, if properly justified, to indicate accomplishment of control of reactivity in the core. If neutron flux is used, the measurements should extend from [$10^{-6}$] to [$10^{+2}$]% of full reactor power. Current value, rate and trend information should be available in the control room [for the primary variable].

C.1.6.1.b - Core Cooling Monitoring

Reactor vessel water level monitoring should be provided to indicate the accomplishment of core cooling. Current value and trend information should be available in the control room.

C.1.6.1.c - Reactor Coolant System Integrity

The measured variables should include reactor pressure, drywell pressure and drywell sump level to indicate the accomplishment and maintenance of Reactor Coolant Pressure Boundary (RCPB) integrity. Current value information should be available in the control room.

C.1.6.1.d - Reactor Containment Integrity

The measured variables should include reactor containment pressure and remote operated containment isolation positions (closed/not closed) to indicate the accomplishment and maintenance of reactor containment integrity. Current value of pressure and valve position status should e available in the control room.

C.1.6.1.e - Radioactive Effluent Monitoring

The measured variable should be noble gas monitoring of the planned release points to indicate accomplishment of radioactive effluent control. The information should be available to the control room operator.

C.1.6.2 - Additional Criteria, Type C Variables

In conjunction with Table 1DA-1, the following should be considered as a minimum number of instruments, and their respective ranges, for accident monitoring instrumentation. In addition, the instrumentation [primary variable] should detect and alarm the following with the current value information in the control room.

C.1.6.2.a - Fuel Clad Barrier Monitoring

These variables indicate a breach in the fuel cladding barriers (i.e., an in-core fuel cladding breach capable of releasing more than 1 percent of fuel cladding gap and plenum activity of the core). The measured variable should be reactor coolant system radioactivity (gross gamma).

Operator sampling of reactor coolant should be used as a means to verify the measured variable. Sampling provisions should permit one sample to be taken every 6 hours with analyses results available within 72 hours of the sampling.

C.1.6.2.b - Reactor Coolant Pressure Boundary Monitoring

These variables indicate a breach in the reactor coolant system that produces a loss of reactor coolant inventory in excess of normal coolant makeup capability. The measured variables used to detect a RCPB breach shall span the full spectrum of design basis accident event break sizes. The measured variable should include drywell pressure and drywell sump level with an accuracy of 20% of span and a response time of less than 10 seconds for input step change of 10%. Current value should be provided in the control room.

C.1.6.2.c - Primary Reactor Containment Pressure Boundary Monitoring

These variables indicate a breach in the Primary Containment Pressure Boundary that is capable of producing radiation releases in excess of Title 10, Code of Federal Regulations.

C.1.6.2.c - Primary Reactor Containment Pressure Boundary Monitoring (Continued)

Part 100 "Reactor Site Criteria" at the exclusion area boundary using TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites, March 23, 1982," source terms.

The measured variable should be primary containment pressure and secondary containment air space radiation monitoring for gross gamma. The containment pressure monitoring instrument range should be the range specified in Table 1DA-1 with a display channel accuracy within ±20% of span and a response time of less than 10 sec for an input step change of 10 percent of span. The secondary containment air space radiation monitoring instrumentation range should be as specified on Table 1DA-1 with a display channel accuracy as specified with a display channel response time of less than 10 seconds for an input step change of ±10% of span. Current value should be provided and alarmed on the control room. Secondary containment air space radiation detectors should respond to gamma radiation photos within any energy range from 60 keV to 3 MeV with an energy response accuracy of ±20% at any specific photon energy from 0.1 MeV to 3 MeV. Overall system accuracy should be within a factor of 2 over the entire range.

C.1.6.2.d - Potential Breach of the Final Fission Product Barrier

The measured variable should be [primary] containment pressure, primary reactor containment hydrogen concentration and RCPB pressure. The primary containment pressure monitoring instrumentation range shoud be as specified on Table 1DA-1 with a display channel accuracy [of] ±10% of span and a response time of less than 1 second for an input step change of ±10% of span. Current value should be provided in the control room. The primary containment hydrogen monitoring instrumentation range should be as specified in Table 1DA-1 with a display channel

C.1.6.2.d - Potential Breach of the Final Fission Product Barrier (Continued)

accuracy of ±10% of span. Initial and subsequent samples should be available at sufficiently short intervals to allow the operator to monitor the value of hydrogen concentration in the containment and take appropriate timely action as required. Current value should be provided in the control room. The RCS pressure monitoring instrumentation range should be as specified on Table 1DA-1 with a display channel accuracy within ±10% of span and a response time of less than 1 second for an input step change of ±10% of span. Current value should be provided in the control room.

Position C.2 - Systems Operation Monitoring and Effluent-Release Monitoring Instrumentation

C.2.1 - Definitions

a. Type D, those variables that provide information to indicate the operation of individual safety systems or other systems important to safety,

b. Type E, those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and in continually assessing such releases.

Interpretation/Implementation Position for Types D & E

Type D and E variables identified in Table 1DA-1 are considered adequate to comply with this position and should be implemented as identified in Table 1DA-1 with the exceptions noted in Table 1DA-1 of this implementation position. Exceptions are bracketed, [ ].

C.2.2

The plant designer should select the appropriate variables and information display channels [from the list of identified variables on Table 1DA-1 and Type D variables] required by his design to enable the control room operating personnel to:

a. Ascertain the operating status of each individual safety system and other systems important to safety to that extent necessary to determine if each system is operating or can be placed in operation to help mitigate the consequences of an accident [(Type D)].

b. Monitor the effluent discharge paths and environs with the site boundary [of his plant] to ascertain if there have been significant releases (planned or unplanned) of radioactive materials and to continually assess such releases [(Type E)].

c. Obtain required information through a backup or diagnosis channel where a single channel may be likely to give ambiguous indication.

C.2.3

The process for selecting system operation and effluent release variables should include the identification of:

a. For Type D -

(1) the plant safety systems and other systems important to safety that could be operating or that could be placed in operation to help mitigate the consequences of an accident; and

C.2.3 (Continued)

(2) the variable or minimum number of variables that indicate the operating status of each system identified in (1) above.

b. For Type E -

(1) the planned paths for effluent release;

(2) plant areas inside buildings where access is required to service equipment necessary to mitigate the consequences of an accident;

(3) on-site location where unplanned releases of radioactive materials should be detected; and

(4) the variables that should be monitored in each located identified in (1), (2), and (3) above.

C.2.4 - Performance Requirements

The performance requirements for system operation monitoring (Type D) and effluent release monitoring (Type E) information display channels should include, as a minimum, identification of:

(1) range of the process variable and monitoring instrumentation [per Table 1DA-1];

(2) required accuracy of measurement [per Table 1DA-1];

(3) required response characteristics, if applicable [per Table 1DA-1];

C.2.4 - Performance Requirements (Continued)

(4) time interval in which the measurement is needed (function time);

(5) the local environment(s) in which the instrument channel components must operate;

(6) requirements for rate on trend information [per Table 1DA-1];

(7) any requirements to group displays of related information; and

(8) any required spatial distribution of sensors.

C.1.5 - Design and Qualification Requirements

The design and qualification criteria for systems operation (Type D) and effluent release monitor [illegible] ype E) instrumentation should be taken from the [appro [illegible] licable] criteria provided in Positions 1.3 and 1.4 for [illegible] 1, 2, and 3 instruments.

Table 1 of this implementation position is considered adequate to comply with the instrumentation and instrument range criteria of Regulatory Guide 1.97 with the exceptions identified in Table 1DA-1 of this implementation position. Exceptions are bracketed, [ ].

Table 1DA-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF 238 NUCLEAR ISLAND

| Variable | System No. | Range | Type | Category | Remarks |
|---|---|---|---|---|---|
| <u>Reactivity Control</u> | | | | | |
| Neutron Flux (value, rate, trend) | C51 | $10^{-6}$% to 100% full power | A,B | 1 | Function detection; accomplishment of shutdown (Primary Coolant) |
| Control Rod Position | C12 | Full in or not full in | B | 3 | Verification [Back-up Variable] |
| Boron Concentration (sample) | D24 | 0 to 1000 ppm | B | 3 | Verification [Back-up Variable] |
| <u>Core Cooling</u> | | | | | |
| Coolant Level in the Reactor (value, trend) | B21 | Bottom of core support plate of [above normal reactor water level] | A,B | 1 | Function detection; accomplishment of mitigation [Primary Variable] |
| | B21 | [Below core plate to above reactor level 1] | B,C | 1 | Potential for breach [Primary Variable] [Alternate to In-core TC] |

Table 1DA-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF
238 NUCLEAR ISLAND (Continued)

| Variable | System No. | Range | Type | Category | Remarks |
|---|---|---|---|---|---|
| Maintaining Reactor Coolant System Integrity | | | | | |
| RCS Pressure (value + alarm) | B21 | 15 psia to 1500 psig ±10%; 1 sec/ 10% response | A,B,C | 1 | Function detection; accomplishment of mitigation. Potential for breach. [Primary Variable] |
| Drywell Sump Level (value + alarm) | E31 | Bottom and top of sump ±20%; 10 sec/ 10% response | B,C | [3] | |
| Drywell Pressure (value + alarm) | T41 | 0 to [30 psig] ±20%; 10 sec/ 10% response | B,C,D | 1,2 | Function detection; accomplishment of mitigation. |
| Primary Containment Area Radiation | - | 0 to $10^7$ R/hr<br>1 to $10^5$ R/hr | E<br>C | 3 | Release assessment<br>Detection of breach |
| Suppression Pool Water Level | P50 | [Top drywell vent to weir wall] | A,C,D | 1,2 | Detection of breach |
| Maintaining Containment Integrity | | | | | |
| Primary Containment Isolation Value Position (Excluding Check Valves) (value) | A11 | Closed<br>Not Closed | B | 1 | Accomplishment of isolation [Primary Variable] |
| Primary Containment Temperature | T41 | 40°F to 250°F | A | 1 | |

Table 1DA-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF
238 NUCLEAR ISLAND (Continued)

| Variable | System No. | Range | Type | Category | Remarks |
|---|---|---|---|---|---|
| Maintaining Containment Inegrity (Continued) | | | | | |
| Primary Containment Pressure (value, rate, trend, + alarm) | T41 | [0 to 30 psig] ±10%; 1 sec/ 10% response | A,B,C | 1 | Function detection; accomplishment of mitigation; potential or actual breach. [Primary Variable] |
| Drywell/Containment Hydrogen Concentration (value) | - | 0 to 30% ±10% | A,C | 1 | Potential for breach [Applicant to provide.] |
| Secondary Containment Area Radiation (value) | - | $10^{-1}$ to $10^{4}$ R/hr 60 keV to 3 MeV; ±20% 0.1 MeV to 2 MeV ±2 x total range; 10 sec/10% response | C,E | 2 | Indication of breach Release assessment [Applicant to provide]. |
| Secondary Containment Noble Gas Effluent | - | $10^{-6}$ μc/cc to $10^{-3}$ μc/cc | C,E | 2 | Potential for breach [Applicant to provide]. |

Table 1DA-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF 238 NUCLEAR ISLAND (Continued)

| Variable | System No. | Range | Type | Category | Remarks |
|---|---|---|---|---|---|
| Maintaining Containment Integrity (Continued) | | | | | |
| Primary Containment Noble Gas Effluent | - | $10^{-6}$ μu/cc to $10^{-2}$ μu/cc | C | 3 | Potential for breach [Applicant to provide.] |
| Suppression Pool Temperature | G51 | 30° to 230°F | A,D | 1,2 | |
| Drywell Air Temperature | T41 | 40° to 440°F | A,D | 1,2 | |
| Fuel Cladding Barrier Monitoring | | | | | |
| Coolant Radiation (value + alarm) | - | 1/2 to 100 times technical specification limit (TSL) ±50%; 5 min/10% response | [NA] | [NA] | [Not needed] |
| Coolant Gamma (1 sample/6 hours) results w/in 72 hr | - | 10 μCi/gm to 10 Ci/gm | C | 3 | Detection of breach |
| System Operation | | | | | |
| Main Steam Line Isolation Valve Leakage Control System [Flow] | E33 | [0 to 110% maximum flow] | D | 2 | [Alternate to pressure] |

Table 1DA-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF
238 NUCLEAR ISLAND (Continued)

| Variable | System No. | Range | Type | Category | Remarks |
|---|---|---|---|---|---|
| System Operation (Continued) | | | | | |
| Containment Spray Flow | E12 | 0 to 110% design flow | D | 2 | [See RHR flow.] |
| Residual Heat Removal (RHR) System Flow | E12 | 0 to 110% design flow | D | 2 | |
| RHR [service water flow] | E12 | [0 to 110% design flow] | D | 2 | [Alternate to heat exchanger temperature.] |
| Low Pressure Coolant Injection System Flow | E12 | 0 to 110% design flow | D | 2 | [See RHR flow.] |
| Reactor Core Isolation Cooling System Flow | E51 | 0 to 110% design flow | D | 2 | |
| High Pressure Coolant [Spray] System Flow | E22 | 0 to 110% design flow | D | 2 | |
| [Low Pressure] Core Spray System Flow | E21 | 0 to 110% design flow | D | 2 | |
| Standby Liquid Control Sytem (SLCS) [Pressure] | C41 | 0 to 110% design flow | D | 2 | [Alternate to flow.] |
| SLCS Storage Tank Level | C41 | Bottom to top | D | [3] | |
| SRV Position | B21 | 0 - 50 psig | D | 2 | |

Table 1DA-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF
238 NUCLEAR ISLAND (Continued)

| Variable | System No. | Range | Type | Category | Remarks |
|---|---|---|---|---|---|
| System Operation (Continued) | | | | | |
| Feedwater Flow | C34 | 0 - 110% design flow | D | 3 | |
| CST Level | - | Bottom + top | D | 3 | Indication of available water [Applicant to provide.] |
| ESF Cooling Water Flow | - | 0 to 110% design flow | D | 2 | [Applicant to provide.] |
| ESF Cooling Water Temperature | - | 32° to 200°F | D | 2 | [Applicant to provide.] |
| High Radioactivity Tank Level | - | Bottom to top | D | 3 | |
| Emergency Vent Damper Position | X73<br>T41<br>X63 | Open/Closed | D | 2 | |
| Standby Energy Status | P53<br>P60<br>P61 | Power, Air | D | 2 | |
| Effluent Monitoring | | | | | |
| SGTS Ventilation Flow | - | [Open/Not Open] | E | 2 | Release assessment |

Table 1DA-1

VARIABLES ASSESSED FOR REGULATORY GUIDE 1.97 ASSESSMENT OF 238 NUCLEAR ISLAND (Continued)

| Variable | System No. | Range | Type | Category | Remarks |
|---|---|---|---|---|---|
| Effluent Monitoring (Continued) | | | | | |
| Other Ventilation Flow | - | [Open/Not Open] | E | [3] | Release assessment |
| Particulate/Halogen Release (sample) | - | $10^{-3}$ μc/cc to $10^{2}$ μc/cc | E | 3 | Release assessment [Applicant to provide.] |
| Environs Radioactivity Monitoring | - | Various | E | 3 | Release assessment [Applicant to provide.] |
| Meteorology | - | Various | E | 3 | Release assessment [Applicant to provide.] |
| Post-Accident Sampling (sample) | - | Coolant, containment air | E | 3 | Release assessment [Sump sample not needed.] |

Table 1DA-2

REGULATORY GUIDE 1.97
SUMMARY OF REQUIREMENTS

CATEGORY 1

| Requirement | | | |
|---|---|---|---|
| 1.3.1 | a. | Whole channel (including indicator) Qualification to Regulatory Guide 1.89, Regulatory Guide 1.100 (equivalent to IEEE 323-1974; IEEE 344-1975) | ______ |
| | b. | Unambiguous Indication with Single Failure | ______ |
| | c. | Station Standby Power | ______ |
| | d. | Available Prior to Accident | ______ |
| | e. | Quality Assurance Applied | ______ |
| | f. | Continuous Indicator | ______ |
| | g. | Variable Recorded | ______ |
| | | (dedicated records if trend essential, otherwise computer memory is permitted) | |
| 1.4 | a. | Isolation Devices for Non-Accident Monitoring | ______ |
| | b. | Identified as to Post-Accident Use (Type A,B,C, only) | ______ |
| 1.5 | a. | Servicing, Testing and Calibration Program | * |
| | b. | Removal from Service Administrative Control | * |
| | c. | Setpoint, Calibration, Testing Administrative Control | * |
| | d. | Human Factors Applied to Type and Location of Display | ______ |
| | e. | Malfunctioning Components Maintainable | ______ |
| | f. | Direct Variable Measurement | ______ |
| | g. | Normal Operation Instrumentation Used | ______ |
| | h. | Testing and Calibration per Regulatory Guide 1.118 | * |

*Applicant to provide.

Table 1DA-2

REGULATORY GUIDE 1.97
SUMMARY OF REQUIREMENTS (Continued)

CATEGORY 2

| Requirement | | | |
|---|---|---|---|
| 1.3.2 | a. | Whole Channel (including display) Qualified to Regulatory Guide 1.89 and Regulatory Guide 1.100 | ______ |
| | b. | High Reliability Power Source (not necessarily station standby) | ______ |
| | c. | Available Per Technical Specifications | * |
| | d. | Quality Assurance Applied | ______ |
| | e. | Individual Display or CRT | ______ |
| | f. | Variable Recorded (NG Effluent and Area Rad only)<br>(dedicated recorder if trend essential, otherwise computer memory) | ______ |
| 1.4 | a. | Isolation Devices for Non-Accident Monitoring | ______ |
| | b. | Identified as to Post-Accident Use (Types A,B,C, only) | ______ |
| 1.5 | a. | Servicing, Testing and Calibration Program | * |
| | b. | Removal from Service Administrative Control | * |
| | c. | Setpoint, Calibration and Testing Administration Control | * |
| | d. | Confusing Indication Minimized; Human Factors Applied | ______ |
| | e. | Malfunctioning Components Maintainable | ______ |
| | f. | Direct Variable Measurement | ______ |
| | g. | Normal Operation Instrumentation Used | ______ |
| | h. | Testing and Calibration per Regulatory Guide 1.118 | * |

*Applicant to provide.

Table 1DA-2

REGULATORY GUIDE 1.97
SUMMARY OF REQUIREMENTS (Continued)

CATEGORY 3

| Requirement | | | |
|---|---|---|---|
| 1.3.2 | a. | High Quality Commercial Grade | ______ |
| | b. | Individual Display or CRT | ______ |
| | | Variable Recorded (NG Effluent, area rad only) | ______ |
| 1.5 | a. | Servicing, Testing and Calibration Program | * |
| | b. | Removal from Service Administrative Control | * |
| | c. | Setpoint, Calibration, and Testing Administrative Control | * |
| | d. | Confusing Indication Minimized; Human Factors Applied | ______ |
| | e. | Malfunctioning Components Maintainable | ______ |
| | f. | Direct Variable Measurement | ______ |
| | g. | Normal Operation Instrumentation Used | ______ |
| | h. | Testing and Calibration per Regulatory Guide 1.118 | * |

*Applicant to provide.

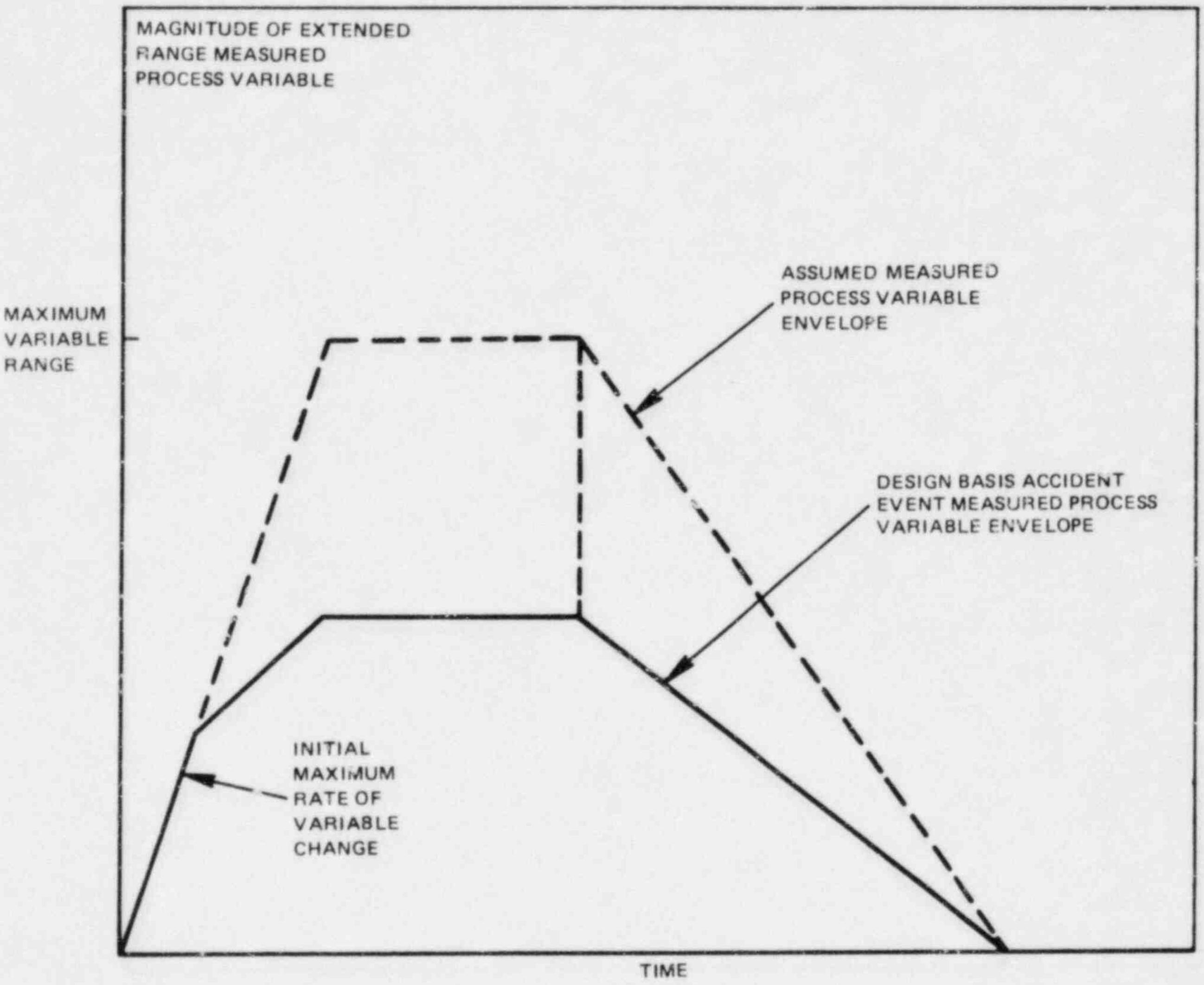


Figure 1DA-1. Typical Environment Qualification Envelope for Type C Instruments

# ATTACHMENT B

# TO APPENDIX 1D

# 238 NUCLEAR ISLAND

# TYPE A VARIABLE ASSESSMENT

# ATTACHMENT B
# TO
# APPENDIX 1D

# 238 NUCLEAR ISLAND TYPE A VARIABLE ASSESSMENT

## 1DB.1 INTRODUCTION

This attachment describes the basis for selecting the Type A variable list used as a basis for the assessment against the provisions of Regulatory Guide 1.97, Revision 2.

## 1DB.2 APPROACH

Regulatory Guide 1.97, Revision 2, defines Type A variables as "Those variables . . . that . . . permit the control room operator to take the specific manually control (s fety) actions for which no automatic control is provided . . . for design basis accident events." The identification of the Type A variables are derived from two sources: GESSAR II, Chapter 15, and the Emergency Procedure Guidelines developed by General Electric for the BWR Owners Group.

## 1DB.3 RESULTS

Chapter 15 contains discussions of numerous events not all of which are design basis accidents. Appendix 15A is a Plant Nuclear Safety Operational Analysis (NSOA) which addresses these same events in the following categories:

Normal Operations
Anticipated Operational Transients
Abnormal Operational Transients
Design Basis Accidents
Special Events

1DB.3 RESULTS (Continued)

Variables associated with normal operations are excluded from further investigation because those activities are planned actions which would not normally be expected to cause a threat to the general public.

Because the Probabilistic Risk Assessment (Section 15D.3) shows that the risk to the general public is dominated by transients rather than design basis accidents, all of the above categories (except normal operations) were considered to determine what parameters required operator action. Tables 1DB-1 through 1DB-4 list the events considered and the primary variables associated with called-for manual action. The manual action variables are taken from either the NSOA or the Chapter 15 event descriptions.

A review of Tables 1DB-1 through 1DB-4 shows that about half of the identified events required no operator action; the safety actions required are automatically initiated. The required manual actions are summarized in Table 1DB-5 along with the associated variables.

The Emergency Procedure Guidelines were also reviewed to determine if there are other variables not specifically identified by Chapter 15 which are associated with required operator actions. Table 1DB-5 includes these additional variables and actions which result from a review of the following guidelines:

RPV Control
Containment Control

Some of these variables, especially those related to emergency action, might be considered beyond the scope of the regulatory guide by virtue of requiring "contingency actions that may also

1DB.3 RESULTS (Continued)

be identified in written procedures." However, they are included with the list of Type A variables because they define conditions which require action on the part of operators to respond to safety-related conditions.

The final list of Type A variables was derived from the variables indicated on Table 1DB-5. Only Reactor Water Temperature ($T_{RPV}$) was deleted since a direct relation between vessel pressure and temperature exists in a boiling water reactor.

Table 1DB-1

238 NUCLEAR ISLAND
ANTICIPATED OPERATIONAL TRANSIENTS

| | Event | Manual Action* Variables |
|---|---|---|
| 7 | Scram | none |
| 8 | Loss of Instrument Air | none |
| 9 | Inadvertent HPCS Pump Start | none |
| 10 | Inadvertent Recirc Pump Start | none |
| 11 | Recirc Flow Control Failure (increasing) | none |
| 12 | Recirc Flow Control Failure (decreasing) | $T_{SP}$, $P_{RPV}$ |
| 13 | Recirc Pump Trip | none |
| 14 | Inadvertent MSIV Closure | none |
| 15 | Inadvertent SRV Opening | $T_{SP}$, $P_{RPV}$ |
| 16,17 | Continuous Rod Withdrawal | none |
| 18 | Loss of Shutdown Cooling | $T_{SP}$, $P_{RPV}$ |
| 19 | Shutdown Cooling Increased Flow | none |
| 20 | Loss of Feedwater | $T_{SP}$, $P_{RPV}$ |
| 21 | Loss of Feedwater Heating | none |
| 22 | Feedwater Controller Failure (maximum demand) | none |
| 23 | Pressure Regulatory Failure (open) | none |
| 24 | Pressure Regulatory Failure (closed) | $T_{SP}$, $P_{RPV}$ |
| 25 | Turbine Trip (w/bypass) | none |
| 26 | Loss of Condenser Vacuum | $T_{SP}$, $P_{RPV}$ |
| 27 | Generator Trip (w/bypass) | none |

Table 1DB-1

238 NUCLEAR ISLAND
ANTICIPATED OPERATIONAL TRANSIENTS (Continued)

| | Event | Manual Action* Variables |
|---|---|---|
| 28 | Loss of On-Site AC Power | $T_{SP}$, $P_{RPV}$, $L_{RPV}$ |
| 29 | Loss of Off-Site AC Power | $T_{SP}$, $P_{RPV}$, $L_{RPV}$ |

*Other than for monitoring $L_{RPV}$ and $P_{RPV}$, securing HPCS/RCIC when RPV level is controlled, transferring mode control switches as appropriate, or verifying protective actions (see Table 1DB-6 for definition of symbols).

Table 1DB-2

238 NUCLEAR ISLAND
ABNORMAL OPERATIONAL TRANSIENTS

| Event | | Manual Action* Variables |
|---|---|---|
| 30 | Generator Trip (w/o bypass) | none |
| 31 | Turbine Trip (w/o bypass) | none |
| 32 | Improper Fuel Loading | none |
| 33 | Recirc Pump Seizure | $T_{SP}$, $P_{RPV}$ |
| 34 | Recirc Pump Shaft Break | $T_{SP}$, $P_{RPV}$ |

*See Table 1DB-1.

Table 1DB-3

238 NUCLEAR ISLAND
DESIGN BASIS ACCIDENTS

| | Event | Manual Action* Variables |
|---|---|---|
| 35 | Control Rod Drop | $T_{SP}$, $P_{RPV}$ |
| 36 | Fuel Handling Accident (outside containment) | None |
| 37 | Loss of Coolant (inside containment) | $H_c^2$, $P_{RPV}$, $L_{RPV}$, $T_{SP}$ |
| 38 | Loss of Coolant (outside containment) | $T_{SP}$, $P_{RPV}$, $L_{RPV}$ |
| 39 | Instrument Line Break (outside drywell) | $T_{SP}$, $P_{RPV}$, $L_{RPV}$** |
| 40 | Feedwater Line Break (outside containment) | $T_{SP}$, $P_{RPV}$, $L_{RPV}$ |
| 41 | Gaseous Radwaste System Leak | ** |
| 42 | Augmented Offgas Treatment System Failure | ** |
| 43 | Radioactive Liquid Waste System Failure | None |
| 44 | Liquid Containing Tank Failures | None |
| 45 | Fuel Handling Accident (inside containment) | None |

*See Table 1DB-1.

**Isolation of line based on channel cross-check, alarm, area radiation, process radiation, area temperature, or leak detection system alarms. Action is for a normal operation shutdown. Isolation is not a required safety action.

Table 1DB-4

238 NUCLEAR ISLAND
SPECIAL EVENTS

| Event | | Manual Action* Variables |
|---|---|---|
| 46 | Shipping Cask Drop | none |
| 47 | Anticipated Transient w/o Scram | $T_{SP}, P_{RPV}, \emptyset$ |
| 48 | Shutdown Outside Control Room | $T_{SP}, \emptyset, P_{RPV}$ |
| 49 | Shutdown w/o Control Rods | $T_{SP}, P_{RPV}, \emptyset$ |

*See Table 1DB-1.

Table 1DB-5

238 NUCLEAR ISLAND
SUMMARY OF MANUAL ACTIONS

| Manual Action | Variable | Source* |
|---|---|---|
| Initiation of Suppression Pool Cooling | $T_{SP}$ | F, E |
| Initiation of Shutdown Cooling | $P_{RPV}$ | F, E |
| Manual Depressurization | $P_{RPV}, L_{RPV}$ | F, E |
| Initiation of $H_2$ Recombiners | $H^2{}_C$ | F |
| Initiation of Leakage Control Systems | $P_{DW}, L_{RPV}$ | F |
| Initiate Standby Liquid Control | $\emptyset, T_{SP}$ | F, E |
| Emergency Action ** if Exceed: | | E |
| Heat Capacity Limit | $P_{RPV}, T_{SP}, L_{SP}$ | |
| Suppression Pool Load Limit | $L_{SP}, P_{RPV}$ | |
| Reference Leg Boiling Limit | $T_C, T_{DW}, T_{RPV}$ | |
| Maximum Drywell Temperature | $T_{DW}$ | |
| Maximum Containment Temperature | $T_C$ | |
| Maximum Containment Pressure | $P_C, L_{SP}$ | |
| Pressure Suppression Limit | $P_L, L_{SP}$ | |
| Initiation of Containment Sprays | $T_C, P_C, L_{SP}$ | E |
| Initiation of Containment Venting | $P_C$ | E |

*E = EPG
F = FSAR

**Scram, Emergency Depressurization and/or RPV Flooding.

Table 1DB-6

DEFINITION OF SYMBOLS

| | | |
|---|---|---|
| $T_{SP}$ | - | Suppression Pool Temperature |
| $T_{DW}$ | - | Drywell Temperature |
| $T_C$ | - | Containment Temperature |
| $T_{RPV}$ | - | Reactor Water Temperature |
| $P_{RPV}$ | - | RPV Pressure |
| $P_C$ | - | Containment Pressure |
| $L_{RPV}$ | - | RPV Level |
| $L_{SP}$ | - | Suppression Pool Level |
| Ø | - | Neutron Flux |
| $H_C^2$ | - | Drywell/Containment Hydrogen Concentration |

ATTACHMENT C
TO APPENDIX 1D

TECHNICAL DESCRIPTION

FOR

ENHANCED LEVEL INSTRUMENT SYSTEM

ATTACHMENT C is PROPRIETARY and is provided under separate cover.