ENCLOSURE 1

SAFETY EVALUATION REPORT INPUT

FOR THE PALO VERDE

SAFETY PARAMETER DISPLAY SYSTEM

1.0 INTRODUCTION

All holders of operating licenses issued by the Nuclear Regulatory Commission (licensees) and applicants for an operating license must provide a Safety Parameter Display System (SPDS) in the control room of their plant. The Commission approved requirements for the SPDS are in Supplement 1 to NUREG-0737 (Reference 1).

The purpose of the SPDS is to provide a concise display of critical plant variables to control room operators. A concise display will aid operators in rapidly and reliably determining the safety status of the plant. NUREG-0737, Supplement 1, requires licensees and applicants to prepare a written Safety Analysis on the SPDS. The Safety Analysis must describe the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents. Licensees and applicants shall also prepare an Implementation Plan for the SPDS. This plan contains schedules for design, development, installation, and full operation of the SPDS as well as a design Verification and Validation Plan. The

Safety Analysis and the Implementation Plan are to be submitted to the NRC for staff review. The results from the staff's review are to be published in a Safety Evaluation Report (SER).

Prompt implementation of the SPDS in operating reactors is a design goal of prime importance. The review of human factors design of the SPDS for operating reactors called for in NUREG-0737, Supplement 1, is designed to avoid delays resulting from the time required for NRC staff review. The NRC staff will not review operating reactor SPDS designs for compliance with the requirements of Supplement 1 of NUREG-0737 before implementation unless a pre-implementation review has been specifically requested by licensees. The staff's review will evaluate the licensee's Safety Analysis and SPDS Implementation Plan. This review will determine if a serious safety question is posed or if the analysis is seriously inadequate. The NRC staff review to accomplish this will be directed at:

(1) confirming the adequacy of the parameters selected to be displayed to detect critical safety functions;

(2) confirming that means are provided to assure that the data displayed are valid;

(3) confirming that the licensee has committed to a human factors program to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator, and

(4) confirming that the SPDS will be suitably isolated from electrical and electronic interference with equipment and sensors that are used in safety systems.

If, based on this review, the staff identifies serious safety questions or seriously inadequate analysis, the Director of IE or the Director of NRR may require or direct the licensee to cease implementation.

2.0 SUMMARY

The staff reviewed Arizona Public Service Company's Safety Analysis Report on the Safety Parameter Display System for Palo Verde Units 1, 2, and 3. Bssed on the results of the review, we conclude that the licensee's Safety Parameter Display System does not fully meet the requirements of Supplement 1 to NUREG-0737. In our safety evaluation, the staff makes several recommendations on human factors features, data validation, parameter selection, etc. Successful resolution of these issues by the licensee should result in a display system that fully meets the requirements of Supplement 1 to NUREG-0737.

3.0 EVALUATIONS

3.1 Background

The Arizona Public Service Company submitted to the Nuclear Regulatory Commission a Safety Analysis Report (Reference 2) on the Safety

Parameter Display System. The staff reviewed the safety analysis and also conducted an audit of the display system. Reference 8 contains a preliminary summary of the audit. The staff was assisted by personnel from Lawrence Livermore National Laboratory during the audit. Appendix A contains Lawrence Livermore National Laboratory's Technical Evaluation Report on the audit.

This Safety Evaluation Report states the results from the staff's review of the safety analysis and the results from our audit of the system.

3.2 Description

The licensee's safety analysis describes the display system. The plant specific system performing the function of the SPDS has been integrated into the emergency response facilities computer system. The SPDS is a computer-based display system consisting of the following major subsystems:

(1) data acquisition system (DAS) in each unit power block;

(2) a Technical Support Center Computer System (TSCCS); and,

(3) display cathode ray tubes (CRT's) and keyboards.

The DAS for each unit at Palo Verde consists of a multiplexer, which transmits, data to the TSCCS for processing. The DAS functions include acquiring raw data, converting data into appropriate units, and transmitting data to the TSCCS. Though the DAS transmits data to the TSCCS, the TSCCS controls all DAS activities. Each DAS configuration includes two computers receiving data from a multiplexer.

The Technical Support Center Computer System (TSCCS) is in the Technical Support Center (TSC). It receives data from the DAS in each of the three units and provides displays of plant parameters via the display CRT's.

The TSCCS consists of two computers and their associated peripheral equipment. Both systems have the same configurations. One processor controls system functions. The other processor is the backup processor operating in standby to provide availability of the overall system in excess of 99%. Shadow memory, automatic CRT switching gear, and interprocessor communications provide for switchover if the active computer fails.

CRT displays are the principal mechanism to present data to the control room operators and TSC/EOF operators during plant normal, abnormal and emergency operation. SPDS bar charts are displayed on these displays. Control Room operators have a graphic terminal in each of the unit's control room. Support personnel have graphic

terminals in the Emergency Operations Facility (EOF), in the Technical Support Center (TSC), and in the satellite TSC adjacent to each unit's control room.

3.3 Parameter Selection

Section 4.1f of Supplement 1 to NUREG-0737 states that:

"The minimum information to be provided shall be sufficient to provide information to plant operators about:

(i) Reactivity control;

(ii) Reactor core cooling and heat removal from the primary coolant system;

(iii) Reactor coolant system integrity;

(iv) Radioactivity control; and

(v) Containment conditions."

For review purposes, these five items are the Critical Safety Functions (CSFs).

The licensee's design of the SPDS draws upon data in the Emergency Operation Procedures (EOP). The information in the safety analysis

describes the EOP as a single procedure. This procedure is applicable to any plant condition that challenges the plant protection system. The EOP uses a three step progression to recover the plant to a stable condition:

Step 1. Maintenance of Safety Functions (SF);

Step 2. Diagnostics to identify the necessary recovery actions; and

Step 3. Performance of recovery actions.

Steps 1 and 2 are performed concurrently to maintain plant safety while the recovery actions are identified. The diagnostic flowchart will identify the recovery actions. If the event cannot be identified, a functional recovery procedure will be used.

The Safety Functions are areas of reactor control. Maintaining these Safety Functions is vital in preventing both core damage and the release of radioactive material into the environment. Also, maintaining these Safety Functions within appropriate limits will mitigate the consequences of an accident. It also provides for an orderly shutdown of the plant should an accident occur.

The licensee's Safety Functions are:

Reactivity Control (RTV);

Heat Removal (HRV);

RCS Inventory and Pressure Control (PIC);

Indirect Radioactive Release Control (IRR);

Containment Integrity (CIN); and

Maintenance of Vital Auxiliaries (VAX).

Our review of the safety analyses' description of these Safety Functions concludes that they exceed the minimum Critical Safety Functions identified in NUREG-0737, Supplement 1.

In the safety analysis, the licensee states that the final parameter selection draws upon Emergency Operation Procedure parameters for the above safety functions. The parameters exclusively selected for display on the SPDS were identified by the licensee's Engineering Department. The parameters were concurred by the Operations Department as being the leading Emergency Operation Procedure's parameters associated with each Safety Function. These parameters were identified to be the most useful to control room operators when assessing the safety status of the plant. This was verified during the SPDS Validation Program. Table I provides a list of these parameters as identified in the licensee's safety analysis.

The staff reviewed the variables selected by the licensee for each of the Critical Safety Functions. We found that the variables selected did provide information needed by operators to evaluate each of the Critical Safety Functions. Our review did note the lack of main steam line radiation as a displayed parameter.

Before isolation, secondary side radiation data provides a rapid assessment of the radiation status for a most likely radioactive release path. The licensee is using the steam generator blowdown radiation parameters and the condenser vacuum exhaust to provide an assessment of plant radiation status in the secondary system. The licensee states that once the the steam generator and/or steamline are isolated, the operator will be able to asses a decrease in radiation on the SPDS. This information will indicate that the steam generator and/or steamline have been isolated and radiation is not being released to the environment. However, for some accident scenarios, after the steam generator and/or steamlines are isolated, radiation may be released to the environment through the atmospheric dump valves and/or steam safety valves. The release may be caused by:

1. operator actions (either intentionally or accidentally);

2. valve failure; or,

3. automatic operation of the valve to mitigate the course of the accident.

The licensee has failed to indicate how radiation releases directly to the environment through the atmospheric dump valves and/or steam safety valves are monitored by the SPDS when the steam generator and/or their steamlines are isolated.

In our audit of the display system, we noted that the addition of Auxiliary Feedwater flow data to the Heat Removal CSF display would be useful. This data serves to provide users an early warning that heat removal via the main steam system is in jeopardy. Also, the data would improve the SPDS user's ability to relate changes in steam generator water level and AFW flow.

In summary, we recommend main steamline radiation be added to the Radiation Release Critical Safety Function or acceptable justification provided for not including this parameter.

3.4 Display Data Validation

The staff evaluated the licensee's safety analysis to determine that means are provided in the display system to assure that the data displayed are valid. From the safety analysis, we learned that data validation was performed by data processing algorithms stored in the computer. The data validation algorithm compares each incoming signal with a like signal of similar range and measurement of the same plant parameter. If the signals deviate from the compared signal by a given amount, the parameter measurement is declared invalid.

During the staff's audit of the SPDS, an examination of the data validation algorithms was conducted. Review of the data validation algorithms identified that the licensee did not develop a consistent basis for interchannel comparison. For example, steam generator water level inputs are valid if they agree within 10 percent.

Review of instrument loop accuracy data showed that under harsh environments, redundant readings could correctly differ by a greater amount.

We recommend the licensee identify and implement interchannel acceptance criteria for data validation. These acceptance criteria should be consistent with the expected deviation between valid inputs under both normal and severe environmental conditions.

3.5 Human Factors Program

The SPDS presents leading indicator parameter data to personnel in the control room. This display is via a set of color monitor displays. To help personnel focus on detailed information, the design contains a set of displays organized into a hierarchy. The highest level in the display hierarchy is deviation bar charts, which are the Safety Parameter Displays (SPDs). The SPDs indicate deviation from the normal key safety parameters so an operator can rapidly assess system deviations with the emergency and recovery operations procedures.

The SPDs consist of a three-level hierarchy of displays: top-level, mid-level, and trend plots. The top-level and mid-level displays are deviation bars that indicate normal or degree of off-normal conditions in the plant. Additional details on these displays are in Appendix A.

The staff's audit of the SPDS identified many human factor discrepancies with the display system. Details on these discrepancies are in Appendix A. Based on our audit findings, we recommend:

- The update rate for deviation bar length and color, and Safety Indicator Block color should be improved;

- The update rate and data resolution for trend displayed data should be based upon process dynamics and operator information needs;

- Invalid data should not be used in the determination of safety function status as displayed by the Safety Indicator Blocks;

- A SPDS user who is normally present in the control room should be identified;

- The licensee should determine if additional and/or periodic retraining on the SPDS is necessary;

- The minor human engineering concerns identified by the staff's audit should be addressed; and

- A standard for display formats should be developed and used so that human engineering problems are avoided during future system modifications.

3.6 Design Verification and Validation

The verification of the SPDS design was performed by constructing the ERFDADS (SPDS) Design Requirements/Capability Matrix. This matrix contains NUREG-0737 Supplement 1 requirements, the guidance capabilities as defined in NUREG-0696, and functional design specification (purchase specification) requirements. This matrix was constructed by the licensee's SPDS Audit Team. This team consisted of representatives from Engineering, Licensing and Operations.

The data in the matrix was used by the licensee's SPDS Audit Team to assist them in verifying the existing SPDS design and its installation against the matrix requirements/capabilities. The following criteria was used to complete the matrix:

- Design Documents (Drawings, Manuals) to verify system design;

- Test Plans (Factory Acceptance Tests "FAT", Operational Availability Demonstration "OAD", Site Demonstration Tests

"SDT", Pre-Operational Test) and results to verify as-built system; and

- Visual Observation to verify as-built system.

The data collected by the licensee's SPDS Audit Team was recorded directly on to the ERFDADS (SPDS) Requirement/Capability Matrix.

The staff's audit of the licensee's SPDS evaluated design verification. Our review found that these activities were conducted in an acceptable manner. Additional details on these activities are in Appendix A.

The method used in validating the SPDS design consists of the "simulator method." The "simulator method" is an actual SPDS that contains transient data to simulate critical safety function violations. This method was determined as being the most effective method for validating the SPDS use because of its' dynamic representation of the safety function groups and associated parameters during the simulated scenarios.

The simulator method consists of completing the following activities:

- designate observer/review team personnel;

- evaluate plant and simulator systems available for simulation;

- select scenarios to be simulated;
- develop data collection techniques for selected scenarios;
- evaluate plant-simulator characteristics;
- select operating crews; and
- conduct crew familiarization.

The validation of SPDS equipment, parameter selection and display organization is outlined below:

1. Use of an SPDS terminal in the Technical Support Center (TSC).

2. Simulation of the following four scenarios:

   TMI Small Break LOCA;

   Back-Up Power Failure;

   Large Steam Line Break; and

   Large Steam Generator Tube Rupture in Steam Generator #2.

3. The Observer/Review team consisted of members from both Engineering and Operations. They presented the simulated scenarios to three licensed personnel knowledgeable in the plant specific Emergency Operating Procedures (EOP) for evaluation.

4. The simulated scenario evaluation consisted of written guides and documented oral interviews.

An in-depth debriefing between the Observer/Review team and operators participating in the scenario was conducted. The debriefing occurred immediately after the completion of the scenario to maximize the effectiveness of the debriefing. Responses from the operators debriefing were documented by the Observer/Review team.

The staff's audit also evaluated design validation. Our review noted some problems with the disposition of Safety Parameter Observations and with the number of modifications to the system. From our analysis of these problems, we recommend:

- Category 2 validation items and Safety Parameter Observations should be entered into the plant change process for tracking;

- Once the SPDS is operational in the plant simulator, a re-validation of the system should be conducted. Simulator

scenarios should be used that exercise all safety functions and integrate the use of the SPDS with emergency procedures and control board instruments.

3.7 Electrical and Electronic Isolation

The licensee's safety analysis did not address the requirement that the SPDS must be isolated from equipment and sensors that are used in safety systems. On January 10, 1986 (Reference 5), a request for additional information (RAI) on isolators was sent to the licensee. The staff also held several telephone conferences (telecons) with the licensee, which resulted in a March 14, 1986 (Reference 6), submittal. The staff evaluated the submittal and forwarded the results of the evaluation to the licensee in a Technical Evaluation Report (TER) dated July 10, 1986 (Reference 7). The TER stated that the Maximum Credible Fault (MCF) test had not been performed on the Energy Incorporated (EI) digital isolation devices. The staff requested that the licensee have the isolators tested for their ability to withstand the MCF test.

In subsequent telecons, the licensee elected to have its Engineering department conduct MCF testing on the electrical isolators. The results of the isolator tests were presented to the NRC SPDS Audit Team during a site audit on November 18-20 1986 (Ref. 4).

The digital isolator test results showed that a barrier breach had occurred in the EI digital isolator during the MCF test. Electrical

energy propagated through the barrier. On this basis, the audit team rejected the digital isolator as a qualified device. The audit team instructed the licensee to disconnect the isolator from the Class 1E circuits.

The licensee appealed the audit team's decision to the NRC management. On December 9, 1986 (Reference 8), the licensee presented additional data and analyses supporting use of the EI digital isolation devices to the NRC staff.

This evaluation addresses the licensee's test program and analyses. The evaluation also addresses the qualification and documentation of the EI digital isolators as acceptable interface devices between Class IE safety-related instrumentation systems and the SPDS.

During the staff's audit, the licensee presented a comprehensive discussion on the electrical isolator test program. Energy Incorporated (EI) analog isolators (Model Nos. 00649-X and 00798-X), with four channels per card, and EI digital isolators (Model 006430), with eight channels per card, are used by the licensee's design. The "-X" in the analog isolator model number indicates the type of input that the isolator accepts, e.g., voltage or current. The digital isolators accept a 0-48VDC input signal supplied through relay contacts.

The licensee performed an analysis of the SPDS signal and power supply circuits to determine the size of the MCF voltage and the

potential current. The analysis assumed that the MCF was applied directly, to the low-voltage power supply input terminals on the non-Class 1E side of the isolator and indirectly to the non-Class 1E signal output terminals.

The single-line diagram supplied with the analysis showed that the isolators could be subjected to a maximum voltage of 120VAC. This voltage was supplied from the ERFDADS distribution panel via 15 amp panel-mounted circuit breakers, which effectively limit the potential fault current to 15 amps. However, in determining the magnitude of the fault current available at the signal output terminals, the licensee assumed that the MCF is applied to the signal output terminals at the data acquisition cabinet rather than at the isolator cabinet. Based on this assumption, a line resistance value of 375 ft. of No. 22 AWG wire (approximately 13 ohms) became part of the MCF current calculation for the signal output terminals, resulting in a source-limited fault.

The staff has not previously considered or accepted a source-limited fault. The staff has stated that the fault must be applied directly to the output terminals of the isolator.

As part of the test program, the licensee applied the MCF voltage/current to the following test configurations:

1. Low-voltage Power supply input terminals on the non-Class 1E side of both the analog and digital isolators;

2. Non-Class 1E signal output terminals of both types of isolators utilizing a 13 ohm resistor between the signal output terminals and the test fixture supplying the MCF; and

3. Non-Class 1E signal output terminals of both types of isolators, bypassing the 13 ohm resistor.

The pass/fail criteria established by the licensee state that there shall be zero leakage current. It also states that the MCF shall not propagate through the barrier from the non-Class 1E side to the Class 1E side of the isolator.

The results of the tests showed that the analog isolators successfully withstood the application of the MCF under the three test configurations and that the pass/fail criteria were met. The digital isolators passed electrical energy from the MCF through the isolation barrier in test configurations 1 and 2, and the pass/fail criteria were not met. However, in test configuration 3, the digital isolators met the pass/fail criteria.

In test configurations 1 and 2 the test instrumentation monitoring the Class 1E input of the digital isolators indicated that 0.5 volts were present on the Class 1E input terminals. These volts identify the leakage of electrical energy through the isolation barrier. Note that leakage of electrical energy through the isolation barrier occurred at the test configurations where the MCF was source-limited

and then applied to the non-Class 1E signal output terminals of the isolator. Furthermore, in test configuration 3, the output circuitry was extensively damaged by the MCF. This circuit damage mitigated the fault and protected the isolation barrier.

The licensee also performed an analysis of the DC circuits that employed the digital isolators. This analysis contained the bounding conditions of the DC circuits. The intent of the analysis was to demonstrate that the breakdown of the digital isolator would have no adverse effect on the safety-related functions of the Class 1E circuits connected to the isolators' inputs. The analysis showed that the digital inputs were dedicated to the SPDS and were not shared by any other system. The analysis also showed that the signals were coming from Class 1E field contacts rated at 120 VAC, 1 ampere and there was contact-to-contact isolation at the contact block. In addition, the circuit's 48 VDC power supply was double-fused, and low-power circuits (48 VDC) were involved.

Because of potential divisional problems among 1E circuits and 1E to non-1E separation problems, the audit team rejected the analysis. At the audit exit meeting, the audit team denied the licensee the use of the digital isolators.

In an effort to have the audit team's decision reconsidered, the licensee requested a meeting with the NRC staff. This meeting

took place on December 9, 1986, at the NRC headquarters in Bethesda, MD.

During the meeting, the licensee demonstrated (by analysis) to the staff that the 0.5v barrier leakage that occurred during the test could not impact the Class 1E circuits. This conclusion is based on had the full MCF voltage of 120 VAC breached the isolation barrier, the isolator's input circuitry would have limited the fault current to approximately 2 milli-amps. The licensee showed that the low power fault (240 milli-watts) could not challenge the field contacts rated at 120 VAC, 1 amp that supply the input signal. The licensee also showed that failure of a digital isolator would not violate divisional boundaries. This conclusion was based on the separation method used at the station in which Divisions A and C do not share isolators with Divisions B and D.

In summary, the licensee demonstrated that a breach of the barrier of a digital isolator would not result in a significant impact on the operation of the Class IE circuits or on the degradation of component life. The licensee also demonstrated that inadvertent operation of any Class 1E circuit would not occur as a result of such a breach.

Based on the review of the test data and the analysis, the NRC staff accepted the digital isolator as a qualified isolator only for the circuit configuration in which it is currently used. The

use of digital isolators in any other circuit will require that they be requalified for the specific circuit configuration in which they are used.

Based on the staff's review of the licensee's submittals with respect to the EI electrical isolation devices, the staff concludes that:

1. The analog isolation device may be used to interface any Class 1E analog circuit with the SPDS;

2. The digital isolation device may be used to interface Class 1E digital circuits with the SPDS only to the extent that the digital circuitry is bounded by this SER (see discussion). The use of these isolators in any other Class 1E circuit will require that they be requalified; and,

3. This equipment, with the limitations noted in 2 above, meets the Commission's requirements in NUREG 0737, Supplement 1.

4.0 CONCLUSIONS

The NRC staff reviewed the Arizona Public Service Company's Safety Analysis to:

1. confirm the adequacy of the parameters selected for display to monitor critical safety functions;

2. confirm that means are provided to assure that the data displayed are valid;

3. confirm that the licensee has committed to a Human Factors Program to ensure that the displayed information can be readily perceived and comprehended so as not to mislead the operator, and

4. confirm that the SPDS is suitably isolated.

Based on its review, the staff confirms that:

1. The SPDS will be suitably isolated from electrical and electronic interference with equipment and sensors that are used in safety systems. However, the staff did place a condition upon the use of certain isolators (see Section 3.7 Electrical and Electronic Isolation for details).

The staff was unable to confirm that:

1. The adequacy of the parameters selected to be displayed to evaluate critical safety functions. We recommend main steamline radiation be added to the Radiation Release Critical Safety Function or acceptable justification provided for not including this parameter.

2. The adequacy of the human factors program in the design of the displays. Our recommendations for this issue are in Section 3.5, Human Factors Program.

Furthermore, we recommend that:

1. the licensee identify and implement interchannel acceptance criteria for data validation;

2. Category 2 validation items and Safety Parameter Observations should be entered in the plant change process for tracking; and,

3. once the SPDS is operational in the plant simulator, a re-validation of the system should be conducted.

Based on these results, the staff concludes that the licensee's SPDS does not fully meet the requirements of Supplement 1 to NUREG-0737. The staff

will evaluate the licensee's response to the above issues and recommendations. The results of this evaluation will be reported in a Supplemental SER.

Appendix A contains a Technical Evaluation Report from the staff's contractor. This report contains additional recommendations, which may be useful to the licensee. One other item identified by the staff's contractor should be included in this list of recommendations. This item concerns a capability for the SPDS to be able to ride through the momentary loss of AC power. Although this capability is currently not a staff requirement, the licensee may find it useful to consider this as a recommendation.

An appropriate implementation schedule will be developed by the Project Manager via discussions with the licensee. Licensees are required to inform the Commission, in writing, of any significant changes in the estimated completion schedule identified in the staff's safety evaluation. The licensee must also inform the Commission when a significant change to the display system has been completed.

5.0 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements, Requirements For Emergency Response Capability," U.S. NRC Report NUREG-0737, Supplement 1, January 1983.

2. Letter from E. Van Brunt, Jr., Arizona Public Service Company, to G.W. Knighton, NRC, Subject: Safety Parameter Display System Safety Analysis Report, dated February 27, 1985.

3. Letter from G.F. Dick, NRC, to Distribution, Subject: Audit of the Safety Parameter Display System, Palo Verde Units 1, 2, and 3, dated December 11, 1986.

4. Letter from E.A. Licitra, NRC, to E.E. Van Brunt, Jr., Arizona Nuclear Power Project, Subject: Safety Parameter Display System - Palo Verde, Units 1, 2, and 3, dated February 5, 1987.

5. Letter from E.A. Licitra, NRC, to E.E. Van Brunt, Jr., Arizona Nuclear Power Project, Subject: Request for Additional Information - Palo Verde Safety Parameter Display System, dated January 10, 1986.

6. Letter from E.E. Van Brunt, Jr., Arizona Nuclear Power Project, to G.W. Knighton, NRC, Subject: Additional Information Concerning Palo Verde Safety Parameter Display System, dated March 14, 1985.

7. Letter from E.A. Licitra, NRC, to E.E. Van Brunt, Jr., Arizona Nuclear Power Project, Subject: Request for Additional Information - Safety Parameter Display System, Palo Verde, dated July 30, 1986.

8. Letter from G.F. Dick, NRC, to Distribution, Subject: Summary of Meeting to Discuss SPDS Testing Program Results, dated December 18, 1986.

TABLE 1

SAFETY FUNCTIONS AND VARIABLES

REACTIVITY CONTROL (RTV)

CEA Position

Log Power

Linear Power

HPSI Flow to RCS

LPSI Flow to RCS

HEAT REMOVAL (HRV)

Sub-Cooled Margin

CET-T Hot

T hot - T cold (Loop 1)

T hot - (Loop 1)

T hot - T cold (Loop 2)

T hot - (Loop 2)

Outlet Plenum Level

SG-1 Level

SG-2 Level

Steam Flow - Feed Flow 1

Steam Flow - Feed Flow 2

PRESSURE & INVENTORY CONTROL (PIC)

Sub-Cooled Margin

Vessel Head Level

RCS Pressure

Pressurizer Pressure

Pressurizer Level

HPSI Flow to RCS

LPSI Flow to RCS

INDIRECT RADIATION RELEASE (IRR)

Plant Vent Stack

Condenser Vacuum Exhaust

Fuel Building Exhaust

S/G 1 Blow Down Radiation S/G 2 Blow Down Radiation Essential

Cooling Water Radiation Control Room Vent Radiation Nuclear

Cooling Water Radiation

CONTAINMENT INTEGRITY (CIN)

Containment Isolation Verification Containment Pressure

Containment Spray Flow

Containment Temperature

Containment Level

Containment Radiation - High Refuel Pool Radiation

H2 Concentration

MAINTENANCE OF VITAL AUXILIARIES (VAX)

HPSI Flow to Loop 1

HPSI Flow to Loop 2

LPSI Flow A to Loop 1

LPSI Flow B to Loop 2

CS Flow A

CS Flow B

Aux. Feed Flow to SG1

Aux. Feed Flow to SG2

Steam Flow - Feed Flow 1

Steam Flow - Feed Flow 2