Enclosure 2

PRE-IMPLEMENTATION AUDIT OF THE SAFETY PARAMETER DISPLAY SYSTEM

FOR THE

CLINTON POWER STATION

January 15, 1985

Prepared for

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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Contract NRC-03-82-096

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FOREWORD

This Technical Evaluation Report (TER) documents the findings from a pre-implementation audit of the Safety Parameter Display System (SPDS) of Illinois Power Company's (IPC) Clinton Power Station. The audit was conducted by a four-man team comprised of two representatives of the NRC's Division of Human Factors Safety, one representative of Science Applications International Corporation (SAIC), and one from Comex Corporation, a subcontractor to SAIC.

The audit consisted of discussions with IPC representatives at Clinton and visits to the Clinton simulator on December 12 and 13, 1984. The SPDS design evolution and present hardware and software features were reviewed. Discussions relevant to each SPDS requirement of NUREG-0737, Supplement 1 were generally structured so that IPC gave a slide presentation on a topic (e.g., SPDS V&V program) and entertained questions primarily regarding points of concern raised by the NRC in its evaluation of IPC's submittals previous to the audit. Visits to the Clinton simulator were conducted to review the SPDS hardware and walkthrough a selected scenario involving the SPDS.

SAIC's participation was provided under Contract NRC-03-82-096. SAIC had not been involved in the review of IPC's SPDS Pre-Implementation Package and the subsequent submittals prior to the audit.

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PRE-IMPLEMENTATION AUDIT OF THE SAFETY PARAMETER DISPLAY SYSTEM FOR THE CLINTON POWER STATION

1.0 INTRODUCTION

This report documents the findings from a pre-implementation audit of the Safety Parameter Display System (SPDS) of Illinois Power Company's (IPC) Clinton Power Station. The purpose of the audit was threefold: (1) to obtain additional information required to resolve any outstanding questions about the SPDS Verification and Validation (V&V) program, (2) to confirm that the V&V program is being correctly implemented, and (3) to audit the results of the V&V activities to date. The requirements set forth in NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," December 1982 (Reference 7) served as the basis of the audit. Due to the absence of the NRC's representative responsible for the review of SPDS electrical or electronic isolation, this requirement was not discussed during the audit.

IPC's human factors review of the SPDS design for Clinton began in 1981 with the development of a display format. In July of 1981 IPC presented the NUCLENET SPDS concept to the NRC. Clinton's process computer system was reviewed by General Physics Corporation during a preliminary design assessment performed in November of 1981. IPC established an "Emergency Response Program Review Team" and with the assistance of a human factors specialist from the University of Illinois, developed and conducted a static checklist review of the SPDS in October of 1983. Presently, IPC has the assistance of Torrey Pines Technology (TPT) in performing a checklist review of the intended SPDS using criteria from industry guidance documents (e.g., NUREG-0700). This second checklist review will be integrated into the Detailed Control Room Design Review (DCRDR) scheduled for completion in June of 1985. A listing of the documents exchanged between the NRC's Human Factors Engineering Branch of the Division of Human Factors Safety and IPC is given as References 1 through 4. The next document to be exchanged will be the NRC's report reflecting the findings of this audit. The findings of the audit follow a brief overview of the background of the SPDS requirements. The SPDS format is presented as an attachment at the end of this TER.

2.0 BACKGROUND

Licensees and applicants for operating licensees are required to provide a Safety Parameter Display System (SPDS). The objective is to "... improve the ability of nuclear power plant control room operators to prevent accidents or cope with accidents if they occur by improving the information provided to them" (NUREG-0660, Item I.D.1). The need for an SPDS was confirmed in NUREG-0737 and in Supplement 1 to NUREG-0737. SPDS requirements in Supplement 1 to NUREG-0737 replaced those in earlier documents. Supplement 1 to NUREG-0737 requires each licensee or applicant to implement an SPDS on a schedule negotiated with the NRC. Human factors guidelines for SPDS design are currently provided in NUREG-0696, NUREG-0835 (draft) and NUREG-0700. The NUREG documents cited are listed as References 5 through 8.

An SPDS is to be established according to the applicant's own safety analysis and implementation plan which must be submitted to the NRC. According to Supplement 1 to NUREG-0737, "the written safety analysis shall include a description of 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." This safety analysis and the specific implementation plan for the SPDS shall be reviewed by the NRC. On-site audits shall be scheduled as necessary to confirm that the applicant is implementing an adequate design program.

The purpose of this Technical Evaluation Report (TER) is to assist the NRC in the technical evaluation process by presenting the findings from the pre-implementation audit of IPC's SPDS for Clinton Power Station. This TER also will provide a basis for constructive feedback to the licensee.

The provisions for SPDS as stated in Supplement 1 to NUREG-0737 can be summarized in terms of the seven elements listed below.

1. Provision of a concise continuous display of critical plant parameters.

2. Location convenient to the control room operators.

3. Incorporation of accepted human factors principles in the design.

4. Procedures for timely and correct safety status assessment.

5. Training for accident response with and without SPDS.

6. Parameter selection sufficient to assess safety status for identified functions.

7. Suitable electrical or electronic isolation.

The audit findings will be formatted in seven sections reflecting the above topics. Each section will include the applicant's proposed design activities, conclusions and recommendations for improvement where necessary.

3.0 PRE-IMPLEMENTATION AUDIT FINDINGS

3.1 Provision of a concise continuous display of critical plant parameters.

Supplement 1 to NUREG-0737 states that "the SPDS should provide a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant." Supplement 1 to NUREG-0737 also states that this system "will continuously display information from which the plant safety status can be readily and reliably assessed by control room personnel who are responsible for the avoidance of degraded and damaged core events."

IPC has developed an SPDS which portrays general plant status, 11 different safety parameters, and containment isolation information all within 34 lines on one CRT. The IPC single CRT SPDS also provides a concise supplementary display of secondary indicators driven by initiation of the alarm system. It appears that IPC has provided a dedicated CRT which serves as a concise means of displaying plant safety status information.

The NRC position concerning continuous display is that all SPDS parameters should be continuously displayed or a method of alerting the operator to changes in the status of SPDS parameters should be provided, such as the critical safety function boxes.

IPC is planning to display some plant safety status information on the 5S CRT on a continual basis. However, all SPDS parameters are not continuously displayed, nor are all SPDS parameters input to the critical safety function boxes. Therefore, IPC appears to have met the provision in Supplement 1 to NUREG-0737 regarding a concise display of critical plant variables but has not fully satisfied the provision for continuous display.

3.2 Location convenient to the control room operator.

Supplement 1 to NUREG-0737 states that "each operating reactor shall be provided with a Safety Parameter Display System that is located convenient to the control room operators." IPC's SPDS CRT is an integral part of the NUCLENET 1000 Control Complex and is located just to the left of the rod control panel. The NUCLENET console functions as the primary plant/operator interface and replaces a significant number of controls and displays required on the traditional benchboard configuration. The SPDS CRT appears generally adequate for seated observation by control room operators. However, the NRC audit team noted that the top of the display is obscured when observed from a standing position directly in front of the SPDS. Except as described in Section 3.6 of this report, the key safety parameters are all available on the 5S CRT to the left of the rod control panel and are therefore convenient to control room operators. The staff noted that a plan exists to perform wiring changes to prevent the operators from moving the SPDS display to an alternative CRT. Since operators may have other displays during certain plant evolutions which are more appropriate for display on the two CRTs closest to the rod control panel, the NRC audit team suggested that IPC consider using dedicated line space on <u>every</u> CRT showing the CSF boxes, rather than dedicating the whole 5S CRT solely to the SPDS function.

3.3 Incorporation of accepted human factor principles in the design.

Supplement 1 to NUREG-0737 states that "the SPDS shall be designed to incorporate human factors principles so that the information presented can be readily perceived and comprehended by the users." IPC is apparently still in the process of conducting a human factors review of the SPDS. The review is to be completed by IPC with the assistance of Torrey Pines Technology in conjunction with the DCRDR.

Documentation of IPC's intent to incorporate accepted human factors principles in the SPDS design is included in its submittal of October 1983. This package contains the initial human factors review of the SPDS. IPC employed design guidance from NUREG-0835 (draft) and NUREG-0700. From these criteria IPC constructed a human factors checklist and tabulated its findings in the 10-page review which contains four major sections; significant concerns, minor concerns, recommendations and unreviewed items. These concerns covered such issues as data validation, visibility of the ARM/PRM displays, radioactivity control data on a separate CRT, and segregation of safety parameters on the display. Other concerns of "lesser significance" included adequacy of color coding, lack of mimics, no indication of flow direction, etc. Several of these concerns identified over a year ago were still unresolved at the time of the NRC audit.

The following paragraphs contain brief summaries of some of the potential problems identified during the audit. For ease of implementation, the NRC's concerns are discussed under headings: (1) SPDS human factors design approach, (2) color coding, and (3) labeling.

3.3.1 SPDS human factors design approach.

The SPDS design has evolved over approximately four years starting with a preliminary display design by an operator in 1981. IPC then presented its concept of an SPDS as part of the NUCLENET system to NRC in July of 1981. IPC submitted a "pre-implementation package" in October of 1983. The human factors design process described in this document apparently was performed by engineers who designed a "strawman" display then looked at the criteria in NUREG-0835 (draft) and NUREG-0700 to see if it fit. The design process does not reflect the necessary top down (safety parameter driven) system function and task analysis activities which would have resulted in an adequate SPDS display format. Furthermore, it appears that although a human factors professional was involved in the development of the assessment checklists, they were applied and interpreted by non human factors personnel. The next step in the SPDS design evaluation process will be taken during the DCRDR supported by Torrey Pines Technology. This will apparently include an EOP walkthrough/talkthrough approach to SPDS and DCRDR validation, the administration of operator surveys, and a checklist review of the SPDS. This effort will commence in July 1985. However, the SPDS may not be operational in time for dynamic evaluation. Overall, the design process was not optimal for the development of an SPDS. The process should have been driven by the safety parameters first, human factors requirements second, and consideration of convenience/ cost last. IPC should commit to an adequate verification and validation process to compensate for its less than optimal design approach. This verification and validation effort must be capable of identifying the need for additional parameters and identifying human factors deficiencies in regard to the manner in which the parameters are displayed. IPC should also commit to implementing the upgrades identified during verification and validation.

3.3.2 Color coding

The basic concern here appears to be an over reliance on the concept of color coding as a method to support the discrimination of information by operators. It appears that the system can generate a limited number of colors (i.e., white, yellow, cyan, red, etc). The use of these colors is not only inappropriate due to the difficulty in detecting the differences in hue but also at odds with the accepted human factors principles concerning the meaning associated with colors. For example, Section 6.5.1.. of NUREG-0700 suggests the use of red to indicate unsafe, danger, immediate operator action required, or an indication that a critical parameter is out of tolerance; yellow to indicate hazard (potentially unsafe), caution, attention required, or an indication that a marginal value or parameter exists; and green to indicate safe, no operator action required, or an indication that a parameter is within tolerance. The SPDS display uses yellow rather than green to indicate a parameter is within tolerance. In addition, green tic marks are used to indicate normal ranges in the bar graph while the numerics which indicate normal readings are yellow. Furthermore, the NRC audit team observed that the hue/saturation of the red alphanumerics do not show up well against the CRT background. This may be aggravated in situations where emergency ambient lighting is used.

In addition to these problems in color coding there are several other concerns which together result in a display which is very difficult to read. The red text has low contrast against the background and the color coding is inconsistent within the display itself. Perhaps the single greatest criticism is the easiest to resolve. There is an over dependence on color coding for information transfer and subsequently there is no redundant (backup) coding scheme to account for partially color blind operators or for SPDS use in a lighting environment other than optimal. Since the colors are limited, hard to distinguish, inappropriate to human factors conventions and inconsistent, perhaps flashing symbols, shape coding, size coding or some other more innovative approach may be more appropriate. It is therefore suggested that alternative approaches to information coding be explored by IPC with help from its human factors consultant.

3.3.3 Labeling

From the NRC discussion with operators during the audit it appears that the use of the letters "I" and "O" as designators of "isolated" and "open" in the containment isolation field of the display are confusing. At least one operator thought the "I" and "O" referred to "inboard" and "outboard."

It is apparent from the preceding discussions that IPC has not fully met the requirement to incorporate acceptable human factors principles. It is strongly suggested that both the analyses which resulted in the parameters selected and the design process which led to the display format be subjected to rigorous diagnostic evaluation by the IPC team supported by the human factors consultant.

3.4 Procedures for timely and correct safety status assessment.

Supplement 1 to NUREG-0737 states that "Procedures which describe the timely and correct safety status assessment when the SPDS is and is not available will be developed by the licensee in parallel with the SPDS." IPC has neither developed nor committed to develop specific procedures describing safety status assessment with and without SPDS. IPC holds that "the SPDS is not a qualified class 1E piece of equipment and thus does not require associated procedures." The IPC position is that proper training on the use of emergency operating procedures (EOPs) and training in the use of the NUCLENET control room will meet the intent of this SPDS requirement. It was not possible to verify the validity of this position during the two day audit. It is recommended that as a minimum IPC personnel incorporate tests of the operators' ability to cope with an unexpected loss of the SPDS during upcoming verification and validation activities.

3.5 Training for accident response with and without the SPDS.

Supplement 1 to NUREG-0737 states that "... operators should be trained to respond to accident conditions both with and without the SPDS available." IPC states that it intends to develop rudimentary training via instructions for SPDS operators. However, those training plans were not ready for presentation at the NRC audit.

3.6 Safety parameter selection sufficient to assess safety status for identified functions.

Supplement 1 to NUREG-0737 states that "the minimum information to be provided shall be sufficient to provide information to plant operators about:

1. Reactivity control
2. Reactor core cooling and heat removal from the primary system
3. Reactor coolant system integrity
4. Reactivity control
5. Containment conditions

The specific parameters to be displayed shall be determined by the licensee."

In an applicable requirement regarding the DCRDR, Supplement 1 to NUREG-0737 states that the review shall consist of "The use of function and task analysis (that had been used as the basis for developing emergency operating procedures) to identify control room operator tasks and information and control requirements during emergency operations. This analysis has multiple purposes and should also serve as the basis for developing training and staffing needs and verifying SPDS parameters."

It appears that the SPDS design philosophy has changed since the last docketing of design information in October 1984 (Reference 3). The original concept treated the area radiation monitor/process radiation monitor (ARM/PRM) display as part of the SPDS. Since then a new critical safety function (CSF) alarm for the ARM/PRM display has been added to the SPDS upper level display. This alarm is actuated by any of the several ARM/PRM alarms associated with the ARM/PRM system. Under the original concept the operator had no direct alarm or display of radiological conditions on the primary SPDS display. While the new concept/design places an alarm directly on the primary SPDS display, the following potential problem exists:

o Radiological parameters are not directly displayed, nor are they directly accessible to the operator. When an ARM/PRM alarm occurs, a second operator must be sent to the ARM/PRM panel about 10 feet away to determine the alarming channel and to obtain parameter values.

In order to address this potential problem IPC SPDS design personnel should evaluate the adequacy of this arrangement during upcoming verification and validation (V&V) walkthroughs of the EOP's, DCRDR and SPDS.

A pre-implementation package submitted by IPC in October of 1983 includes the SPDS verification and validation team report on human factors. Based on a close inspection of these documents and the findings of the NRC audit it appears that neither the selection nor operational definition of the safety parameters was based on any formal top down system function and task analysis. In addition the team that developed the pre-implementation package, although multidisciplinary, had no input from human factors professionals. There appears to have been no a priori integration of human factors criteria into the parameter selection process.

During the course of the audit the NRC audit team received and reviewed numerous documents and presentations concerning verification and validation work performed on the SPDS design project. However, all of this work was oriented toward the SPDS hardware and software operability and reliability. None of the work appeared to emphasize the identification of operator information and action needs as they relate to identifying and assessing the safety status of the plant. The following sections identify specific problems and areas requiring further investigation with respect to parameter selection and display.

3.6.1 Radioactivity Release

Although a radioactivity release (control) CSF alarm block has recently been added to the SPDS display, the following problems still exist:

- o Current design does not transmit drywell high radiation monitor output to the ARM/PRM panel and therefore will not actuate the radiation control CSF alarm.

- o A plan exists to add plant vent stack noble gas concentration instrumentation to the ARM/PRM panel. Vent stack flowrate is already available on the ARM/PRM panel. Since technical specifications, emergency plan classification guides (EPIP on EALS), EOP entry conditions, etc. are all written in terms of release rates instead of concentrations, the SPDS designers should consider developing a simple algorithm to display release rate directly. This would eliminate the need for operators to make the hand calculation to determine the relationship of release rate to the various action statements in the procedures referenced above.

- o None of the ARM/PRM parameters were selected for direct display on the SPDS. With the change in philosophy which excludes the ARM/PRM panel from being part of the SPDS, the designers should evaluate the benefits of adding key radiological parameters such as containment radiation and stack release rate directly to the SPDS display.

- o IPC's SPDS design team demonstrated only a cursory knowledge of the new radiological monitoring equipment being installed in the plant. The design team should add this expertise for the remainder of the implementation phase of SPDS.

3.6.2 Containment ΔP

Secondary containment ΔP (Combustible gas control volume to outside atmosphere) does not trigger the containment integrity CSF alarm. The design team should consider adding this parameter as a trigger point to the existing containment integrity CSF or adding a separate CSF for secondary containment (leaving the existing CSF dedicated to drywell and primary containment). Note that Revision 3 of the GE emergency procedure guidelines treats primary and secondary containment control as separate guidelines. The SPDS design team contended that secondary containment ΔP units on the SPDS of PSID was correct. Upon further investigation by the NRC audit team, it was shown that the proper units are inches of water. Errors such as this must be corrected prior to the final installation stage of the project.

3.6.3 Reactivity

The power control (reactivity control) CSF is triggered only by the upscale average power range monitors' (APRMS) trip at 108% of the CSF. As a minimum, it should also be triggered by a signal indicating valid reactor protection system (RPS) trip with failure to achieve a downscale (< 3%) APRM trip within a few seconds. This is the entry condition for the ATWS emergency procedure guideline. Failure to evaluate and include such features may be due to the fact that no formal system function and task analysis was conducted during the SPDS design process.

3.6.4 Coolant Control

The reactor coolant system integrity CSF alarm is triggered by only one parameter: drywell floor drain sump flow. This parameter is provided to the SPDS from a single, non 1E instrument which monitors the coolant level in a V-notch located in a Weir upstream of the sump pump. Therefore, the <u>sole</u> input to the reactor coolant system integrity CSF cannot be subjected to any kind of confidence check. Other parameters should be evaluated as possible redundant indicators of failure of the reactor coolant system. Possibilities include safety relief valve position, reactor vessel level and drywell temperature. The present design does not provide for a CSF alarm associated with a break in an interfacing system outside the drywell. Addition of the suggested parameters as triggers to the CSF would provide indication of the interfacing LOCA situation.

3.6.5 Group Isolation

The existing SPDS display for group isolations is triggered only by a successful closure of all valves in the isolation group. A demand signal for an isolation is not indicated. Questioning of the SPDS team and available operators did not confirm that positive indication of the conditions warranting a group isolation exist elsewhere in the control room. The SPDS design team should evaluate the benefits of including group isolation demand signals on the SPDS in addition to the current successful isolation indication provided.

3.6.6 Containment Pressure

Primary containment pressure (outside drywell, inside primary containment) does not trigger the containment integrity CSF alarm. This is probably the primary indicator of abnormal conditions in the primary containment and yet was not included in the CSF alarm logic.

The above examples demonstrate the need to utilize the task analysis results and V&V process being developed for the EOP and DCRDR project for the final parameter selection and SPDS design activities. Should IPC personnel identify SPDS deficiencies during the DCRDR, the findings and their resolutions should be reported to the NRC. IPC personnel stated that the SPDS is to be operational just prior to the submission of the DCRDR summary report. The SPDS related HEDs should be included as a separate section of the DCRDR summary report.

3.7 Suitable electrical and electronic isolation.

Supplement 1 to NUREG-0737 states that "The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems." The NRC audit team did not include an I&C specialist and therefore did not evaluate the final test results for the TEC model 2200 isolation devices being used to isolate SPDS signals from class 1E safety equipment. IPC personnel committed to submit results of this testing to the NRC for evaluation by specialists in this field (Re: GDC 24, APP A, 10 CFR 50).

4.0 SUMMARY OF CONCLUSIONS AND RECOMMENDATIONS

It is the general conclusion of SAIC that the IPC SPDS does not meet the provisions for SPDS development contained in NUREG-0737, Supplement 1. Although IPC does indicate a commitment to provide a concise, continuous display of safety status information to support rapid and accurate operator response to an accident, it does not appear to have a sufficient understanding of the requirements at this time to implement that commitment. The following constructive critiques and recommendations are provided in summary form for each of the SPDS provisions.

4.1 Concise continuous display.

To ensure that the plant safety status information will be continuously displayed, IPC should consider (1) incorporating into the design a continuous display of the critical safety function boxes which includes input of <u>all</u> SPDS parameters as well as <u>direct</u> <u>access</u> to the underlying parameter values, or (2) continuous display of <u>all</u> SPDS parameters on a dedicated CRT.

4.2 Location convenient to operator.

SPDS may not be visible to a standing operator and may be fixed to one specific CRT in order to support the provision for "continuous display." IPC should consider (1) a means to reduce glare and still allow observation by a standing operator and (2) not establish the 5S CRT as the only location SPDS information can be displayed.

4.3 Incorporation of accepted HFE principles

SPDS design approach in general and color coding and labeling specifically are areas of non-compliance with accepted human factors principles. IPC personnel together with substantial support from human factors consultants should subject the design process and display format to rigorous diagnostic evaluation with regard to human factors principles. IPC should commit to the implementation of changes which enhance operator ability to rapidly and accurately respond to off-normal sequences.

4.4 Procedures for safety status assessment.

IPC contends the SPDS specific operating procedures are not required. IPC should test operator ability to use SPDS information and to cope with SPDS outages during upcoming V&V activities. If specific procedures are demonstrated to be necessary then IPC should comply.

4.5 Training for accident response with and without SPDS.

Rudimentary training/orientation instructions and exercises should be developed to assure effective SPDS use.

4.6 Parameter selection.

IPC has not conducted a formal SFTA in support of parameter definition, selection, or verification. Without a priori knowledge of operator information requirements it is not likely to ensure the necessary parameters in an adequate display format. IPC should subject parameter selection and information presentation to rigorous evaluation during the joint SPDS review and DCRDR.

4.7 Electrical and electronic isolation.

There was no evaluation of this provision during the NRC review. IPC will submit pertinent information to NRC specialists for assessment.

4.8 Miscellaneous findings.

- o Only wide range reactor vessel water level is supplied to the SPDS. Due to lack of time and lack of knowledge by IPC personnel, it was not possible to ascertain the adequacy of this range of indication during all accident conditions. IPC personnel should review the adequacy of the level instrumentation with respect to operation during elevated drywell temperatures and while controlling level to control power during the ATWS event.

- o Numerous parameters used in the SPDS do not undergo a confidence check because they are measured by a single channel or by parallel channels of the same parameter. IPC personnel should evaluate alternative means of validating data such as rate of change, comparison to average, etc.

- The provisions for a manual alarm acknowledge and for reflash of SPDS CSF alarms are presently in the conceptual stage of design. The SPDS design team should meet and agree on the exact hardware and features to be installed.

REFERENCES

1. Clinton Power Station SPDS and supporting displays design document, Revision 2, December 1984.

2. Clinton Power Station Unit 1, SPDS Verification and Validation Plan, October 1983.

3. Clinton Power Station Unit 1, response to NRC requests for additional information on SPDS, October 1984.

4. Clinton Power Station Unit 1, SPDS functional description, February 1984.

5. NUREG-0660, Vol. 1, "NRC Action Plan Developed as a Result of the TMI-2 Accident," USNRC, Washington, D.C., May 1980; Rev. 1, August 1980.

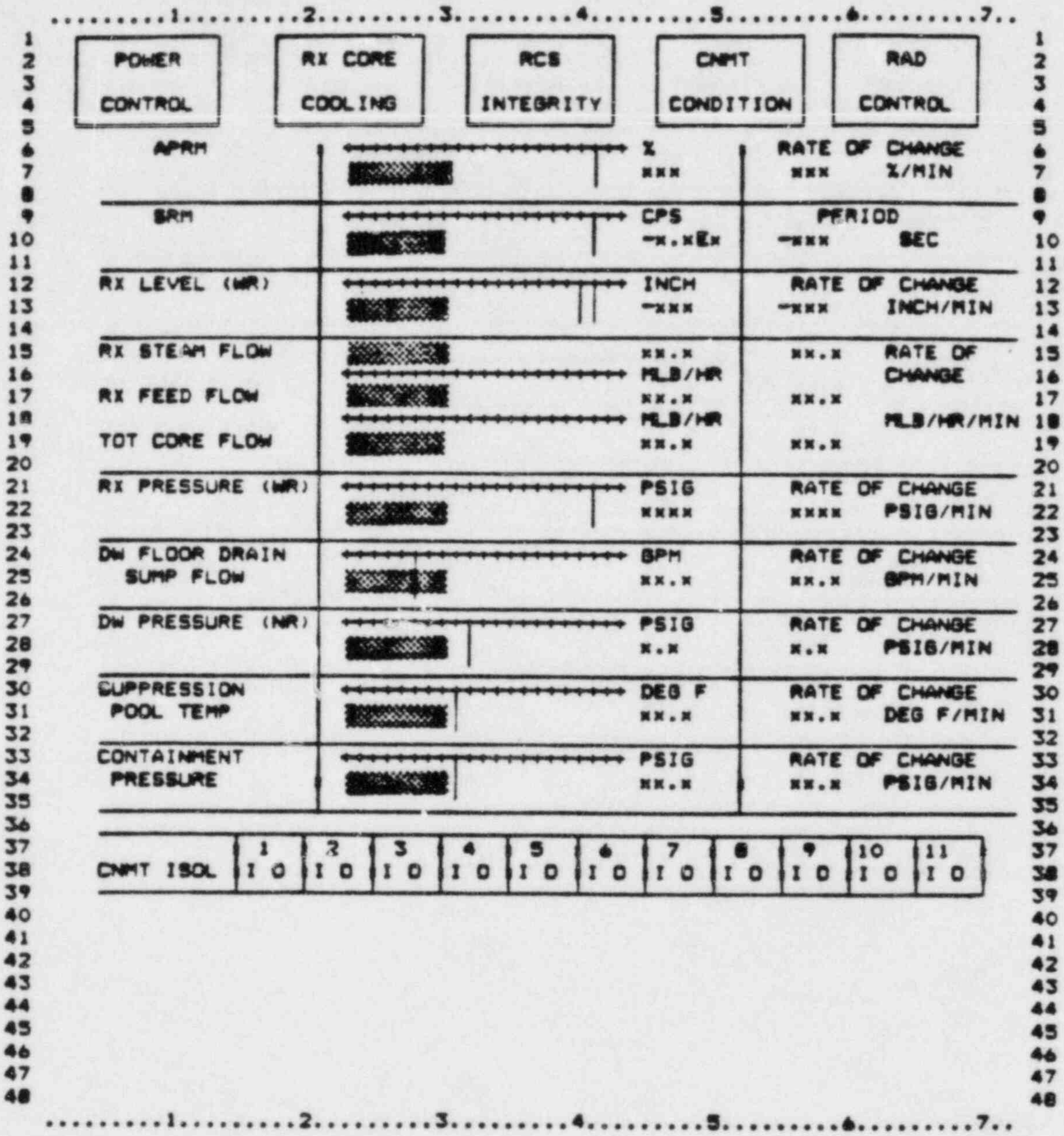
6. NUREG-0737, "Requirements for Emergency Response Capability," USNRC, Washington, D.C., November 1980.

7. NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," USNRC, Washington, D.C., December 1982, transmitted to reactor licensees via Generic Letter 82-33, December 17, 1982.

8. NUREG-0700, "Guidelines for Control Room Design Reviews," USNRC, Washington, D.C., September 1981.

9. NUREG-0835 (draft), "Human Factors Acceptance Criteria for the Safety Parameter Display System," USNRC, Washington, D.C., October 1981.

ATTACHMENT



SS SPDS PERMANENT DISPLAY

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.........1.........2.........3.........4.........5.........6.........7..

   RX WTR LVL         -xx.x IN          SUPP POOL LVL     xx.x FT
   DW PRESS            xx.x PSIG        SUPP POOL TEMP   xxx.x F
   DW TEMP            xxx.x F           CNMT PRESS        xx.x PSIG
   SRV STATUS (OPEN/CLOSED)             CNMT TEMP        xxx.x F
   DW FL SUMP FLOW    xx.x GPM          CNMT/DW H2 CONC  xxx.x%   xxx.x%
   SDV A/B LEVEL     xx/xx GAL          SEC CNMT ΔP  x.xx x.xx x.xx PSID

.........1.........2.........3.........4.........5.........6.........7..
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ALARM INITIATED DISPLAY

CLINTON POWER STATION UNIT #1
SAFETY PARAMETER DISPLAY SYSTEM
NRC PRE-IMPLEMENTATION AUDIT
ENTRANCE MEETING

* Introductions by NRC Staff & Illinois Power
* Entrance Briefing by NRC Staff
* Program Overview and Current Status by Illinois Power
    - SPDS Verification & Validation Program
    - SPDS Parameter Set Selection & Validation Process
    - SPDS Design Development & Implementation Process
    - SPDS Availability Calculations
    - SPDS Human Factors Reviews

TLR

SPDS PARAMETER SET SELECTION AND VALIDATION PROCESS

* SPDS Parameter Set Definition:

  Minimum Set sufficient to determine plant in safe condition.

  - Limits Bulk of Information
  - Provides Sufficient Information

* Synoptic Set of Parameters

  - Concerned with Safety Status at Present

SLIDE #1 TLR

SPDS PARAMETER SET SELECTION AND VALIDATION PROCESS

CRITICAL SAFETY FUNCTIONS (CSFs)

* Overall Function - Containment of Radioactivity.

* BARRIER INTEGRITY
  - Fuel Cladding
  - Reactor Coolant System
  - Primary Containment
  - Secondary Containment

* HEAT TRANSPORT
  - Fuel Cladding
  - Reactor Coolant System
  - Primary Containment

* REACTIVITY CONTROL

SPDS PARAMETER SET SELECTION AND VALIDATION PROCESS

VALIDATION PROCESS

* Reviewed CSFs Against Clinton SPDS Parameter Set
  - Categorized Each Parameter by the CSF it Monitors.
  - V&V Team Evaluated Appropriateness of Parameters.

* Plant Transient & Accident Review
  - FSAR, Chapter #15 Analysis
  - WASH 1400 : Reactor Safety Study
    - Loss of All Decay Heat Removal
    - ATWS
  - Various Misc. Events Chosen by V&V Team.

* Comparison of CPS SPDS Parameter Set to Other Accident-Monitoring Lists for BWRs
  - NSAC/21
  - Regulatory Guide 1.97
  - BWR Generic Emergency Procedure Guidelines (EPGs)
  - NUREG/CR-1440

SPDS PARAMETER SET SELECTION AND VALIDATION PROCESS

VALIDATION RESULTS

* Overall Monitoring of CSFs Comprehensive.

* Very Close Coorespondence With EPGs - Particularly Usefull to the Plant Operator(s).

* V&V Team Recommendations

  - SPDS Parameter Set Additions

    - Secondary Containment d/P

    - SRV Position Status

    - Suppression Pool Temperature also on Permanently Displayed Horizontal Bar Graph.

  - SPDS Parameter Set Deletions

    - Reactor Feed Flow

    - Reactor Recirculation Flow

    - Drywell Equipment Sump Flow.

  - All Recommendations Implemented With Exception of Deleting the Reactor Feed Flow Parameter.

SPDS PARAMETER SET SELECTION AND VALIDATION PROCESS

CURRENT STATUS OF SPDS DISPLAY

* Parameter Set Validation Report Completed (submitted IP Letter U-0676, dated 10/28/83.

* V&V Team Recommendations Have Been Implemented with the Exception of Rx Feed Flow Parameter.

* NRC Concerns Related to CSF Overview Status

  - Identified During April 5, 1984 IP Presentation to Staff.

  - CSF "Status Boxes", Including "Radioactivity Control", added to Top of SPDS Display.

  - SPDS "5S" Display, Permantly Displayed on NUCLENET CRT #5, Now a Complete and Stand-alone Status of All CSFs.

  - Per IP Letter U-0745, dated 10/2/84, ARM/PRM Display Panel No Longer Part of CPS SPDS.

* No Additional Work Remains for the SPDS Parameter Set Selection.

SLIDE #5 TLR

SPDS VERIFICATION AND VALIDATION PROGRAM

PLAN DESCRIPTION

* Modelled After NSAC/39 - NRC Finds Acceptable Generic Plan.

* Verification

  - Review of Requirements (Both NRC and System Functional).
  - Review of Design to Ensure Requirements are being Implemented.

* Validation

  - Detailed Design Review
  - Test & Evaluation of Integrated Hardware & Software System(s).
  - Ensure Developmental Problems are Identified and Resolved.

* V&V ACTIVITIES

  - System Requirements Review
  - System Design Review
  - Field Walkdowns
  - Field Verification Testing
  - Validation Testing
  - V&V Documentation.

SLIDE #1 TLR

Figure 1.0

CPS VERIFICATION & VALIDATION PROGRAM

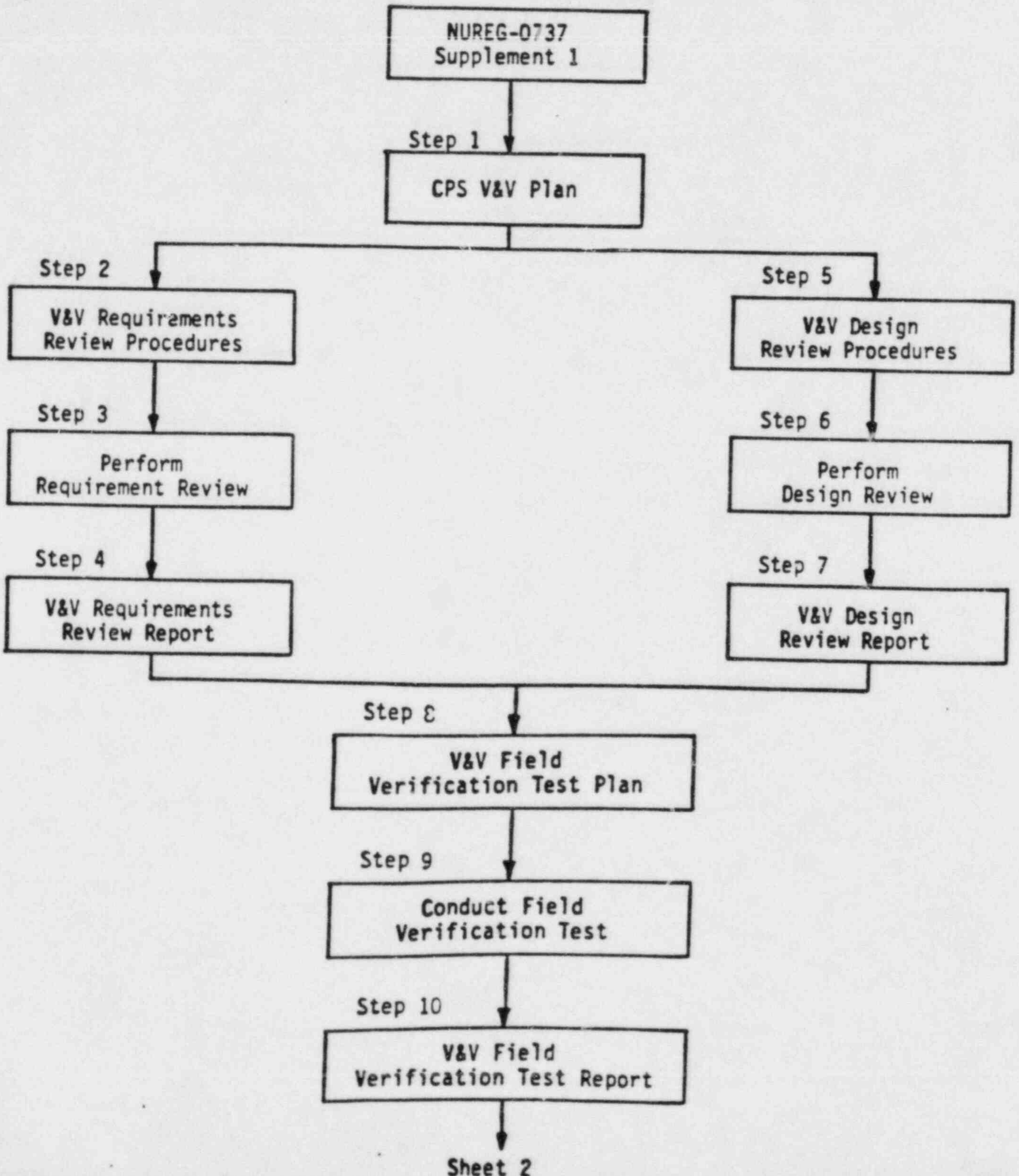
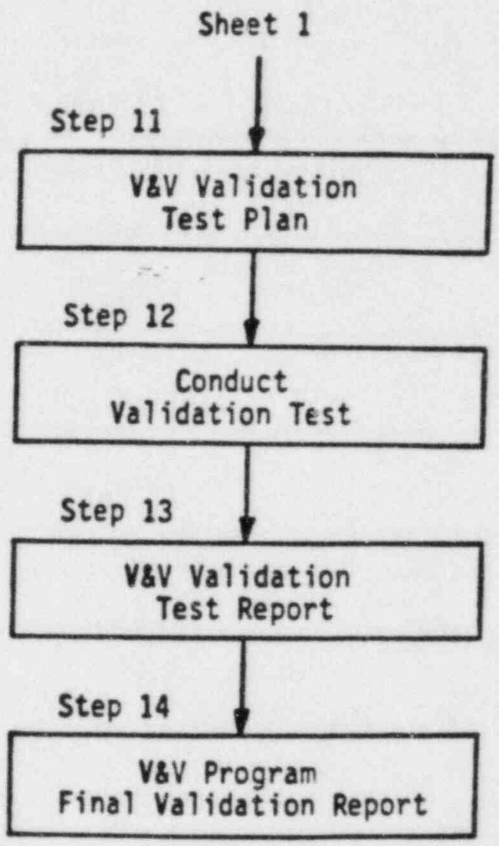


Figure 1.0


Sheet 1
Step 11
V&V Validation Test Plan
Step 12
Conduct Validation Test
Step 13
V&V Validation Test Report
Step 14
V&V Program Final Validation Report

SPDS VERIFICATION & VALIDATION PROGRAM

CURRENT V&V STATUS

* "Clinton Power Station Safety Parameter Display System Parameter Set Validation Report" –

  – Submitted to NRC in SPDS Pre-Implementation Package via IP Letter U-0676, dated 10/28/83.

* "Clinton Power Station Verification and Validation Plan for Safety Parameter Display System" –

  – Implemented October 1983.

  – Submitted to NRC in SPDS Pre-Implementation Package via IP Letter U-0676, dated 10/28/83.

* V&V "SPDS Requirements Review Report" –

  – Completed and Issued on April 24, 1984.

* V&V "SPDS Design Review Report" –

  – Completed and Issued on C tober 31, 1984.

SPDS VERIFICATION AND VALIDATION PROGRAM

CURRENT V&V STATUS (continued)

* V&V "SPDS Test Plan" -
  - General Plan Complete (draft).
  - Field Verification Test and Validation Test Checklists are Drafted and Under Review.
  - Software Test Requirements Document Developed (draft).
  - Software Test Procedures (draft) Being Finalized.
  - Startup C&IO and Pre-Operational Phase Test Procedures (draft) Being Finalized.
* V&V Testing Scheduled for May-July 1985 Time-frame.
* Final V&V Program Results Report Scheduled for Issue July 1985.

SLIDE #5 TLR

SPDS DESIGN DEVELOPMENT PROCESS

* July 1981 NUCLENET Owner's Group Presentation to NRC Staff.

* SPDS Display Originally Developed by CPS Plant Operations Staff.

* Detailed Design Document
  - Revision 0 : 10/10/83.
  - Revision 1 : 12/5/83.
  - Revision 2 : 12/84.

* Requirements Document
  - Revision 0 : 9/20/83.
  - Revision 1 : 12/84.

* Software/Hardware Functional Description
  - Prepared from SPDS Design Document Revision 1 and SPDS Requirements Document Rev. 0.
  - Sent to NRC Staff via IP Letter U-0695, dated 2/10/84.

SLIDE #1 TLR

SPDS DESIGN DEVELOPMENT PROCESS

* IP Presentation to NRC on SPDS Development on April 5, 1984 at NRC Bethesda, Maryland Offices.

  - Briefing Books Provided

    - SPDS Pre-Implementation Pkg. Material.
    - SPDS Requirements Document, Rev. 0.
    - SPDS Software/Hardware Functional Description.
    - Results of SPDS Availability Study.
    - Color Photos of Control Room Display Panels.

* IP Responses to NRC Questions Provided via IP Letter U-0745, dated 10/2/84.

* Additional NRC Concerns on Optical Isolation Devices Used at CPS have been Identified - Under Review by IP.

SLIDE #2 TLR

SPDS DESIGN DEVELOPMENT PROCESS

SPDS AVAILABILITY STUDY

* Adequacy of CPS SPDS : NUREG-0696 Availability Criteria

  - Operational Unavailability Goal of 0.01 When Reactor Above Cold Shutdown Status.

  - Unavailability Goal of 0.2 During Cold Shutdown.

* IMPELL Contracted to Perform Work.

* Fault Tree Analysis Methodology

  - Mean Time To Repair (MTTR) and Mean Time Between Failure (MTBF) Calculated:

$$\text{UNAVAILABILITY} = \frac{\text{MTTR}}{\text{MTTR} + \text{MTBF}}$$

  - Sources of Data

    - IEEE Standard 500

    - Reactor Safety Study

    - Military Handbook MIL-HDBK-117D.

SLIDE #3 TLR

SPDS DESIGN DEVELOPMENT PROCESS

SPDS AVAILABILITY STUDY (continued)

* **Scope of Evaluation**

  - Those Portions of PMS/DCS Process Computer System Required to Operate NUCLENET CRT #5.

  - Did Not Include:

    - External SPDS Displays (e.g. those to be in EOF or TSC).

    - SPDS Data Links.

    - Software/Firmware Induced System Failures.

* **Four (4) Cases Evaluated**

  - CASE #1 :
    - System as Designed & Conservative Failure Rate Estimates.

  - CASE #2 :
    - System as Designed & Most Likely Failure Rate Estimates.

  - CASE #3 :
    - Replacement of DCS/PMS Common Drum with Large Core Storage Device / Conservative Failure Rates.

  - CASE #4 :
    - Replacement of DCS/PMS Common Drum with Large Core Storage Device / Most Likely Failure Rates.

SPDS DESIGN DEVELOPMENT PROCESS

Summary - SPDS Unavailable Results

(With Plant Operating)

	Conservative Cases	Most Likely Cases
SPDS WITH DRUMS	.081	.033
SPDS WITH BULK MEMORY INPLACE OF COMMON DRUM	.0199	.0037

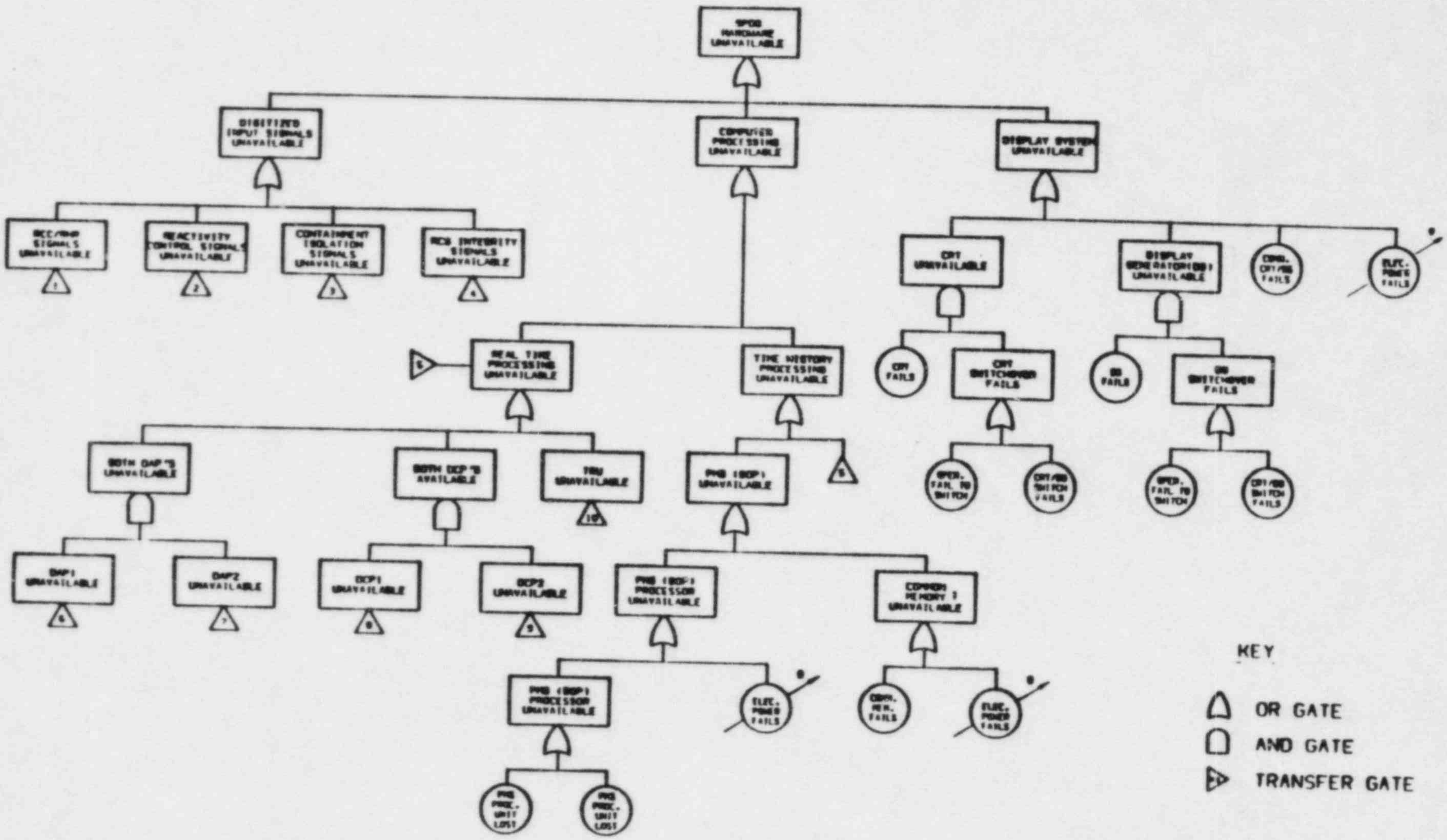
(With Plant in Cold Shutdown)

* With Respect to Unavailability Goal of 0.2

  - Review of Process Computer Operational History Indicated This Not Expected to be a Problem.

SLIDE #5 TLR

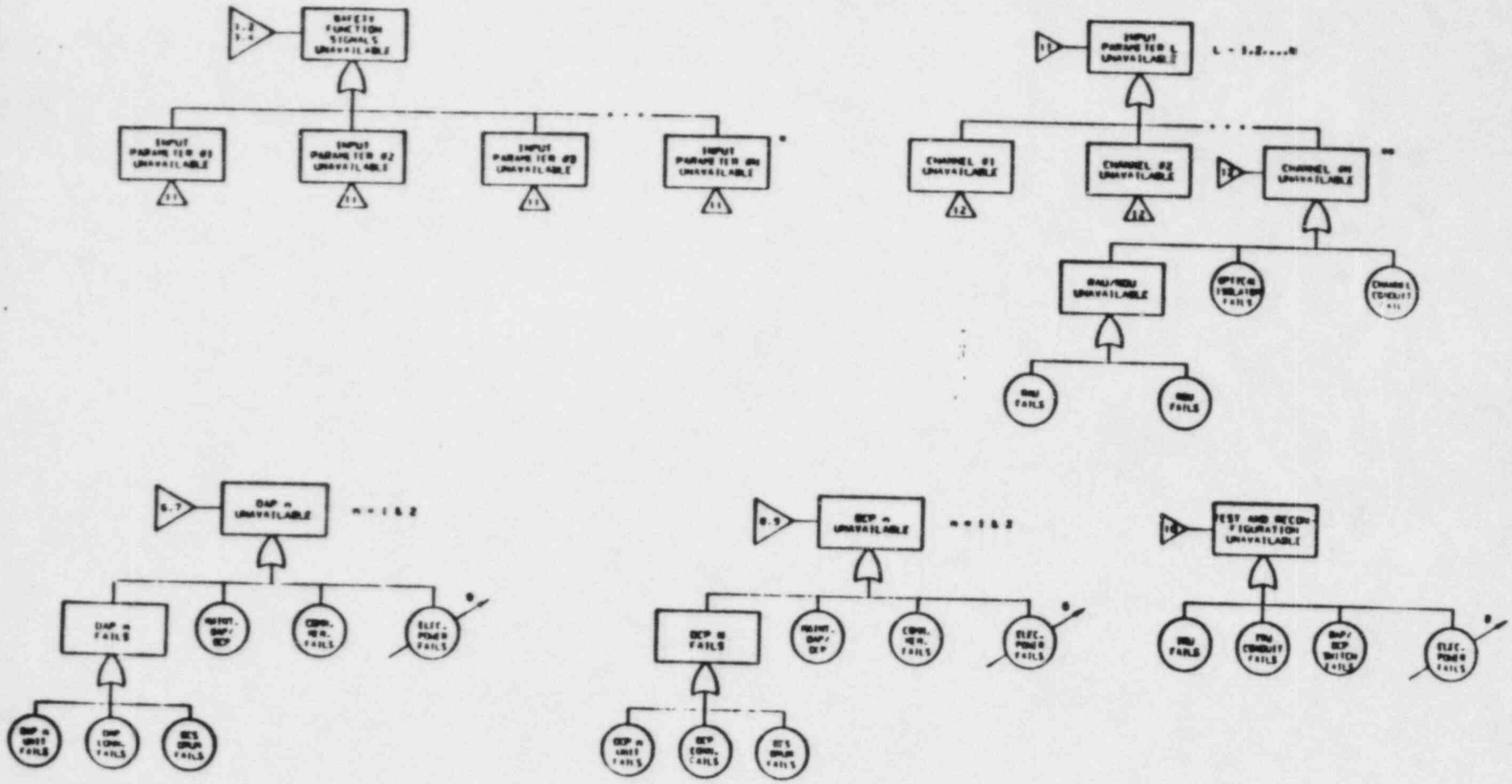
FIGURE 4-1
FAULT TREE MODEL FOR SPDS UNAVAILABILITY



SHT. 1 OF 2

FIGURE 4-1

FAULT TREE MODEL FOR SPDS UNAVAILABILITY



NOTES:

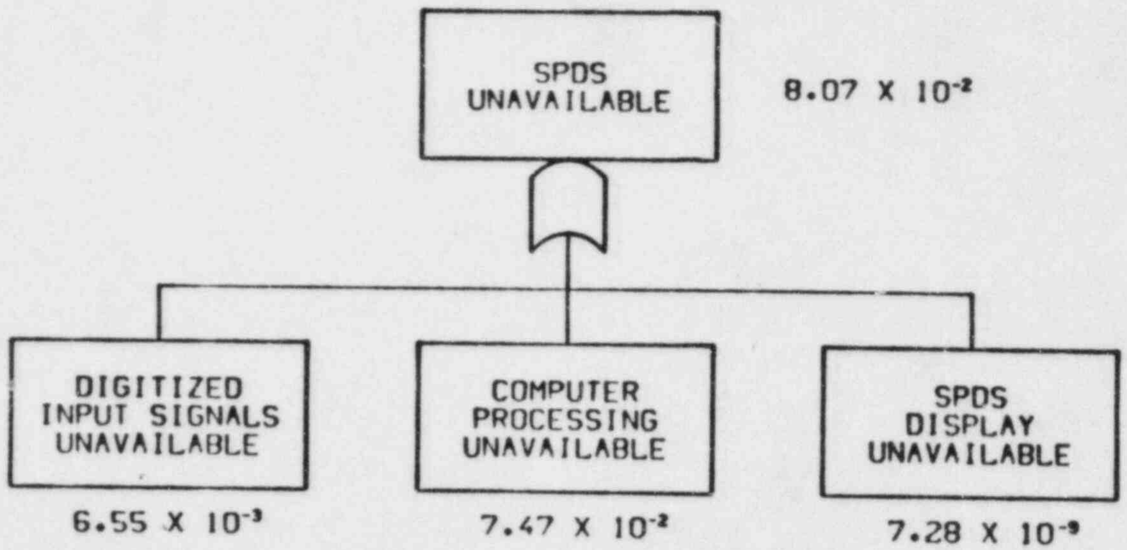
* EACH SPDS DISPLAY CATEGORY CONSISTS OF N INPUT TYPES

** EACH INPUT TYPE CONSISTS OF N CHANNELS

SHT. 2 OF 2

IMPELL

FIGURE 4-2A
SPDS UNAVAILABILTY:
CONSERVATIVE FAILURE RATE CASE



- CHANNEL CN 112
  (3.36 X 10^{-3})
- RAU 16
  (2.24 X 10^{-3})
- RDU 8
  (9.6 X 10^{-4})

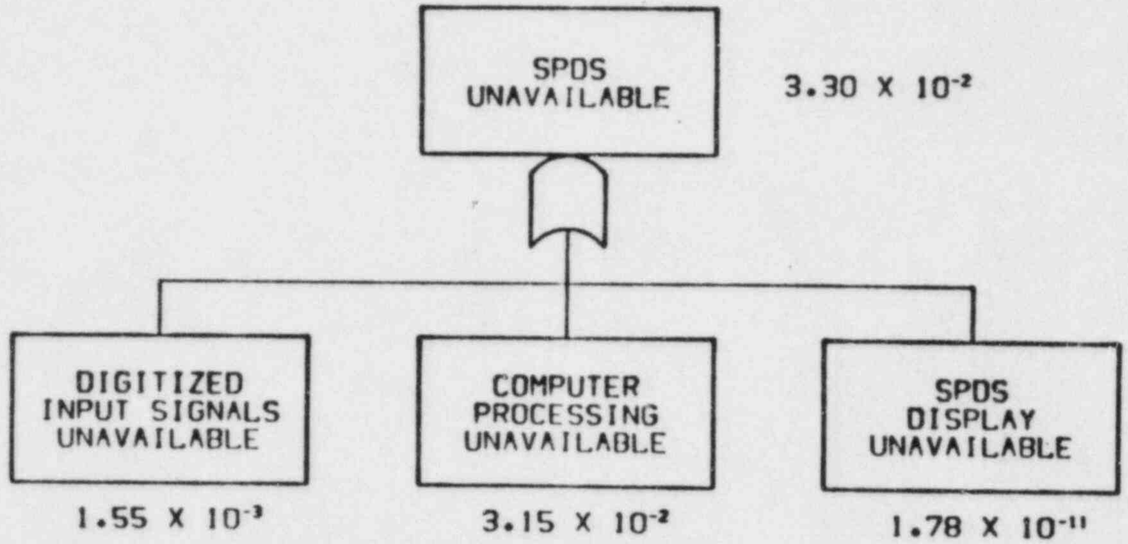
- DCS DRUM
  (3.2 X 10^{-2})
- PMS DRUM
  (3.2 X 10^{-2})
- TRU CN 100
  (3.0 X 10^{-3})

- CRT CN3, CRT SWITCH
  (5.04 X 10^{-9})
- CRT CN3, OPERATOR ERROR
  (1.13 X 10^{-9})
- DISPLAY GEN., CRT SWITCH
  (4.82 X 10^{-10})

FIGURE 4-2B

SPDS UNAVAILABILTY:

MOST LIKELY FAILURE RATE CASE



- RAU 16 (8.64×10^{-4})
- RDU 8 (3.68×10^{-4})
- CHANNEL CH 112 (3.14×10^{-4})

- DCS DRUM (1.48×10^{-2})
- PMS DRUM (1.48×10^{-2})
- MAINTENANCE (1.1×10^{-3})

- CRT CN3, OPERATOR ERROR (6.49×10^{-12})
- DISPLAY GEN., OPTR. ERROR (3.56×10^{-12})
- CRT, OPERATOR ERROR (3.56×10^{-12})

FIGURE 4-3A
SPDS UNAVAILABILTY:
CONSERVATIVE FAILURE RATE CASE
WITH DCS AND PMS DRUMS REPLACED

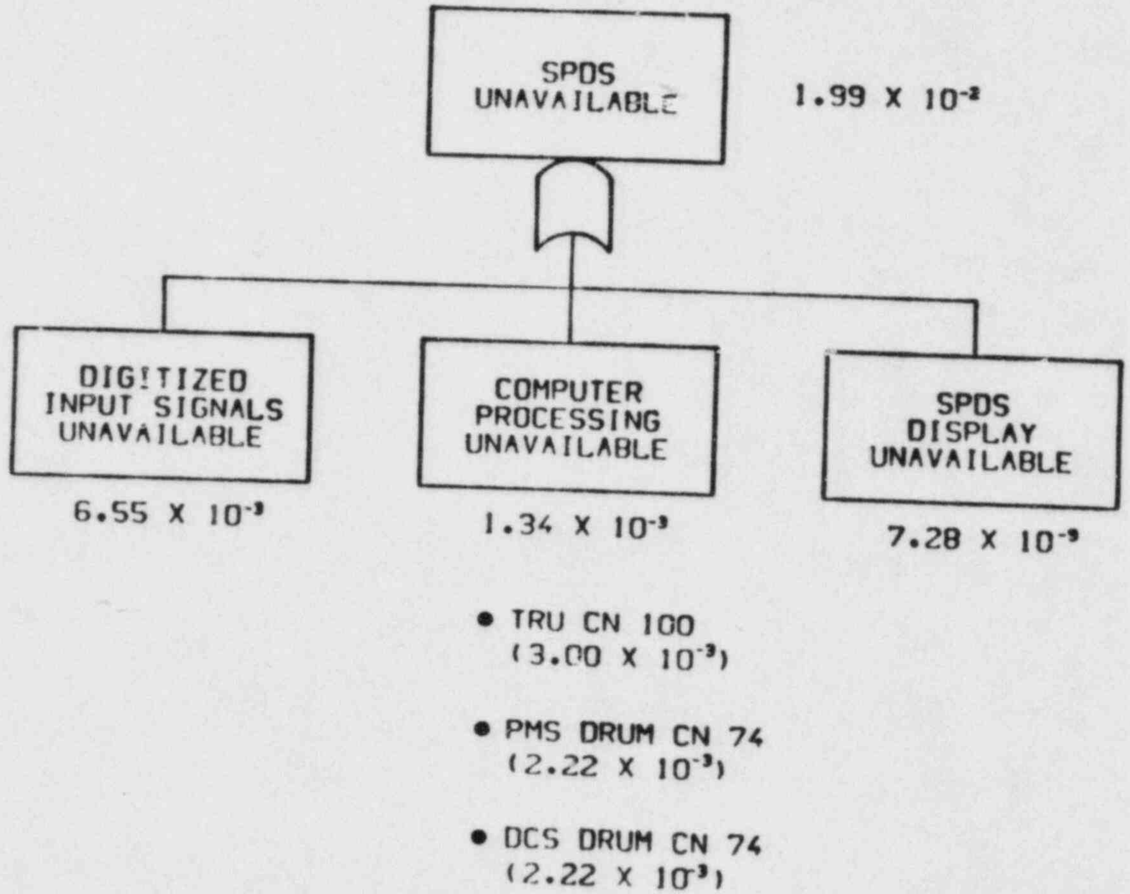
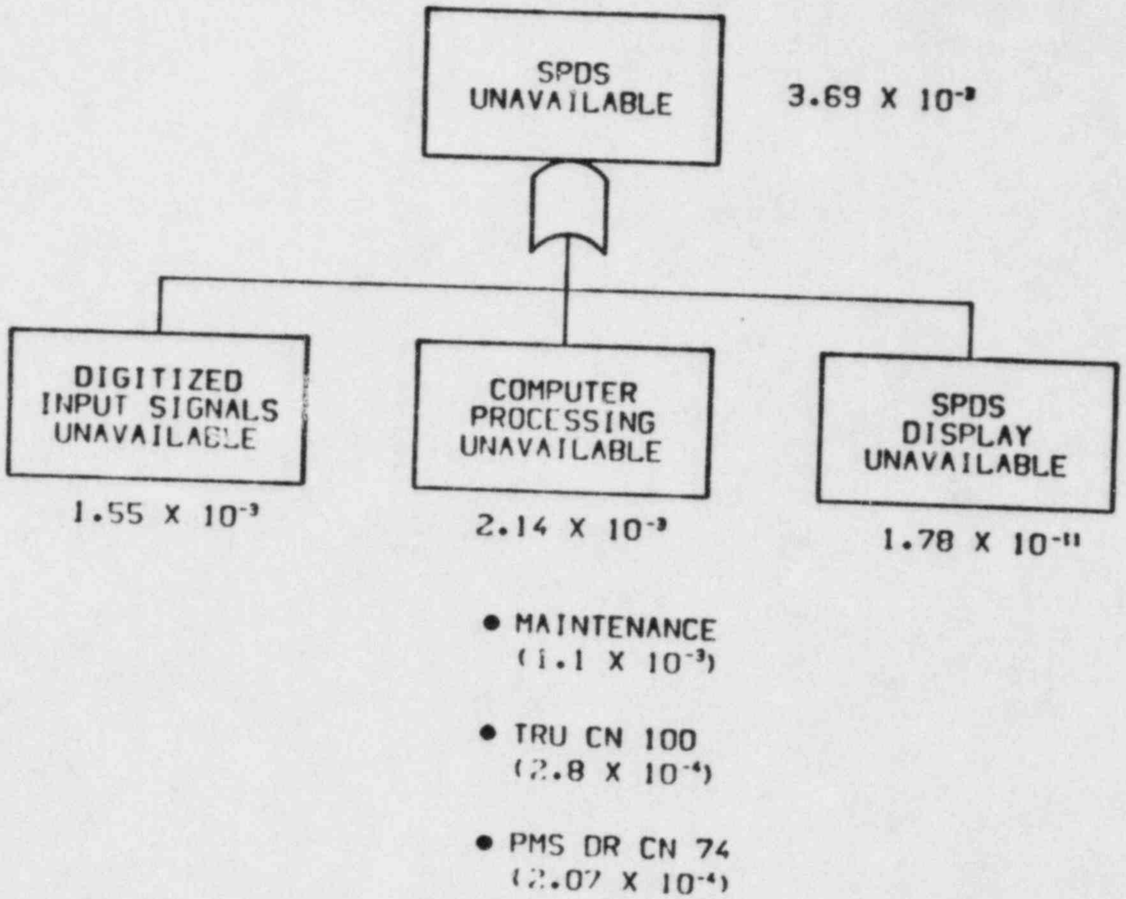


FIGURE 4-3B
SPDS UNAVAILABILTY:
MOST LIKELY FAILURE RATE CASE
WITH DCS AND PMS DRUMS REPLACED



SPDS DESIGN DEVELOPMENT PROCESS

SPDS DESIGN STATUS

* Completing Final Design Implementation - i.e. Plant Process Computer System Software Being Debugged and Made Operational.

* SPDS Software Installed in DCS System.

* Testing of DCS/SPDS Software Underway.

* Documentation:

  - SPDS Requirements Document Rev. 1 Approved.

  - SPDS Design Document Rev. 2 Approved.

  - Software Test Requirements Document (Complete Draft).

  - SPDS Test Procedures Still Under Development.

SPDS DESIGN DEVELOPMENT PROCESS

SPDS DESIGN STATUS (continued)

* **Optical Isolation Devices**

- IP Responses to NRC Questions Currently Under NRC Review.
- Technology for Energy Corp. (TEC) Isolator Cabinets Completely Assembled and Tested
- Cabinets will be Shipped from TEC Following IP Quality Assurance Approval of Fabrication Records.
- Estimated Shipment Date is December 15, 1984.

* SPDS Design Completion Scheduled for Early 2nd Quarter 1985.

* CPS Operator Training to be Completed Prior to Fuel Load.

SLIDE #13 TLR

SPDS DESIGN DEVELOPMENT PROCESS

DATA VALIDATION

DCS Operating System Validation

* Reasonableness Test: each channel is validated as it is digitized, against a data base range.

  A. Real Point (associated with a Plant sensor)

  B. Psuedo Point (value depends upon a real point) will be turned WHITE if any of its source points:

  - Are Deleted from Processing
  - Failed the Reasonableness Test
  - Out of Scan
  - Undefined
  - Has an Inserted Value

  C. Real/Psuedo Point

  - Has an Inserted Value

* Color WHITE Used to Display LOW CONFIDENCE Data.

  - Means Sensor is No Longer Active for any one of the Above Reasons
  - Last Good Value of Data is Displayed in WHITE.
  - WHITE Color will Flag Operators as to the Uncertainty of the Data and to Regard the Point Values Accordingly.

SPDS DESIGN DEVELOPMENT PROCESS

DATA VALIDATION

* Digital Points

  -- Group Failures.

Redundant Data Point Comparison

* Drywell Pressure (NR) Only

* Displays PID B21DA008 and Confidence Checks with PID B21DA009.

* For VALID Data:

  — /DWP−DWi/ $\leq$ Lambda * DWP

    where Lambda = Scaling Factor (will be determined when the source transducer and channel accuracy have been measured.)

* For INVALID Data:

  — Displayed Value is Turned WHITE but Continues to Display the Value of DWP.

  — /DWP−DWi/ < Lambda * DWP

SPDS DESIGN DEVELOPMENT PROCESS

DATA VALIDATION

Averaging Algorithm

* Parameters:

  - Average Power Range Monitor (APRM)
  - Source Range Monitor (SRM)
  - Wide Range Reactor Water Level
  - Supression Pool Temperature

* APRM: Average of C51DA021, C51DA022, C51DA023, and C51DA024 after a Confidence Check.

* SRM: Average of C51DA001, C51DA002, C51DA003, and C51DA004 after a Confidence Check.

* Wide Range Reactor Water Level:

  Average of B21DA002, NB-DA401, NB-DA402, and NB-DA403 after a Confidence Check.

* Suppression Pool Temperature:

  Average of CMBA001, CMBA002, CMBA003, and CMBA004 after a Confidence Check.

SPDS DESIGN DEVELOPMENT PROCESS

DATA VALIDATION

Averaging Algorithm

* For VALID Data:

$$X' = \frac{1}{N} \cdot \sum_{i=1}^{N} X_i \quad \text{average value}$$

$/X - X_i/ < \text{Lambda} * X$

* For INVALID Data:

  - Value is Turned WHITE but Continues to be Displayed.

    $/X - X_i/ > \text{Lambda} * X$

    The Value of Lambda will be Determined When the Source Transducer and Channel Accuracy have been Determined.

HUMAN FACTORS REVIEW

Overview

* Human Factors Review Conducted in October 1983 by an IP Interdisciplinary Review Team and Dr. Charlie Hopkins (U. of I. Human Factors Specialist).

  Used Draft NUREG-0835 and NUREG-0700 Checklists.

* Results of Review Submitted to NRC Staff in November 1983 as an Enclosure to SPDS Pre-Implementation Package (IP Letter U-0676).

* NRC Staff Questions Addressing Human Factors Concerns Sent to IP in Letter dated August 17, 1984.

* CPS Responses to these Questions Provided via U-0745 dated October 2, 1984.

HUMAN FACTORS REVIEW

Present Status

* CPS Detailed Control Room Design Review (DCRDR) Integrates the EOP V&V and the Human Factors Review of SPDS.

* Man-in-the-Loop Testing:

  Involves Operator Testing Using CPS Emergency Operating (Off Normal) Procedures and

  - CPS Simulator and/or

  - Main Control Room Walkthroughs

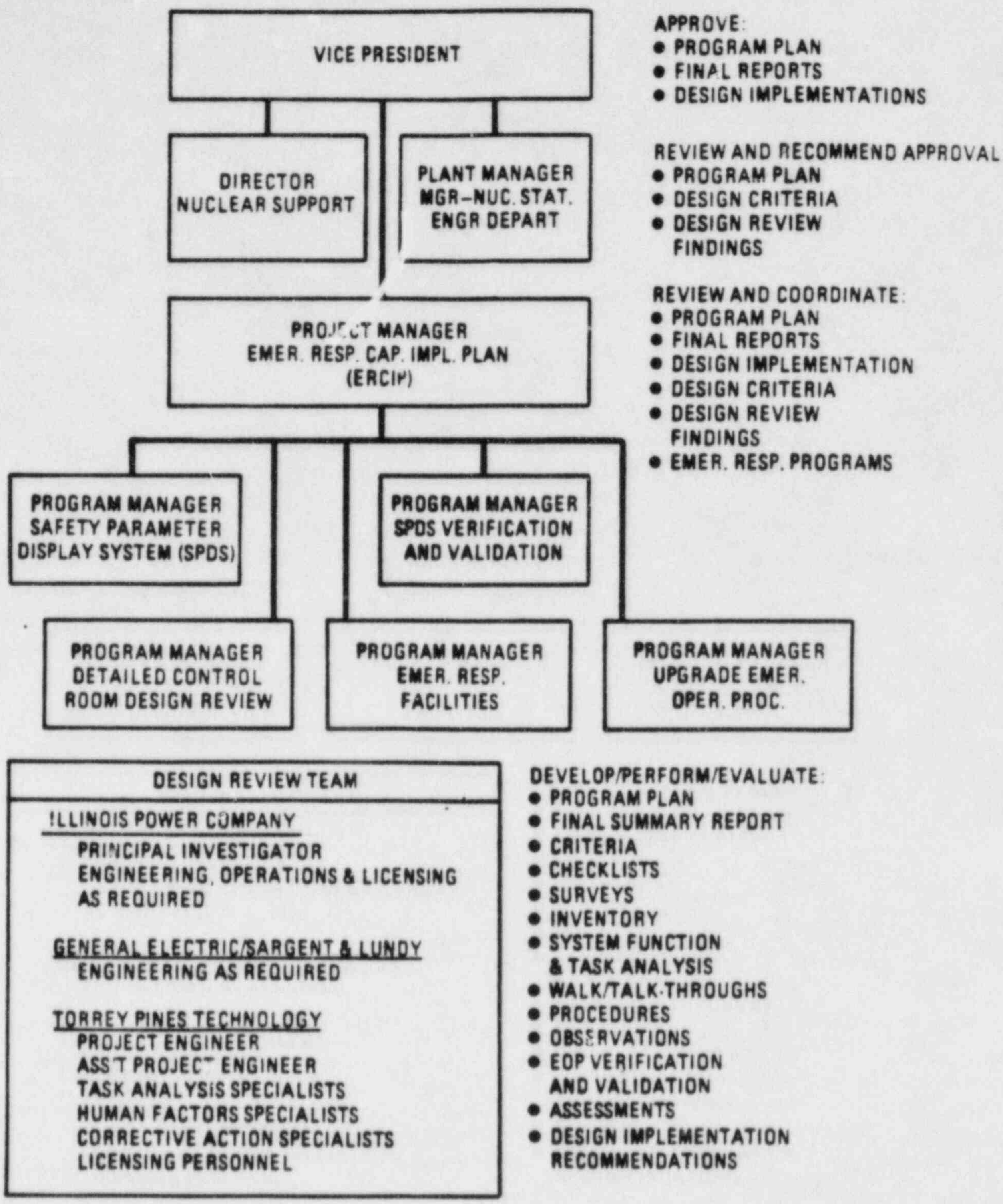
  Evaluates Operator Interface with SPDS during Simulated Plant Accidents/transients.

Schedule

* CPS DCRDR Scheduled to be Complete by May 1985.

* Final DCRDR Report to NRC Staff by July 1985.

SLIDE #2 TLR



Overall ERCIP Organization

SPDS HUMAN FACTORS REVIEW

Background

- The original SPDS was developed by a CPS assistant shift supervisor who was SRO Certified. The operator was a system specialist on the process computer.

- The concept of integrating SPDS on Nuclenet was presented on NRC Staff in July 1981 by the Nuclenet Owners Group.

- The process computer was reviewed for human factors during the NRC Staff's Preliminary Design Assessment in November 1981.

- IP interdisciplinary human factors review team utilized draft NUREG-0835, NUREG-0700 and industry guidance. Checklist was made and the display was reviewed against the checklist in a method similar to the Preliminary Design Assessment. The review team consisted of:

  - Controls & Instrumentation Engineers
  - Electrical Engineers
  - Computer Specialists
  - SRO Certified Operators
  - Nuclear Engineers
  - Human Factors Specialist - Dr. Charles O. Hopkins

Present Status

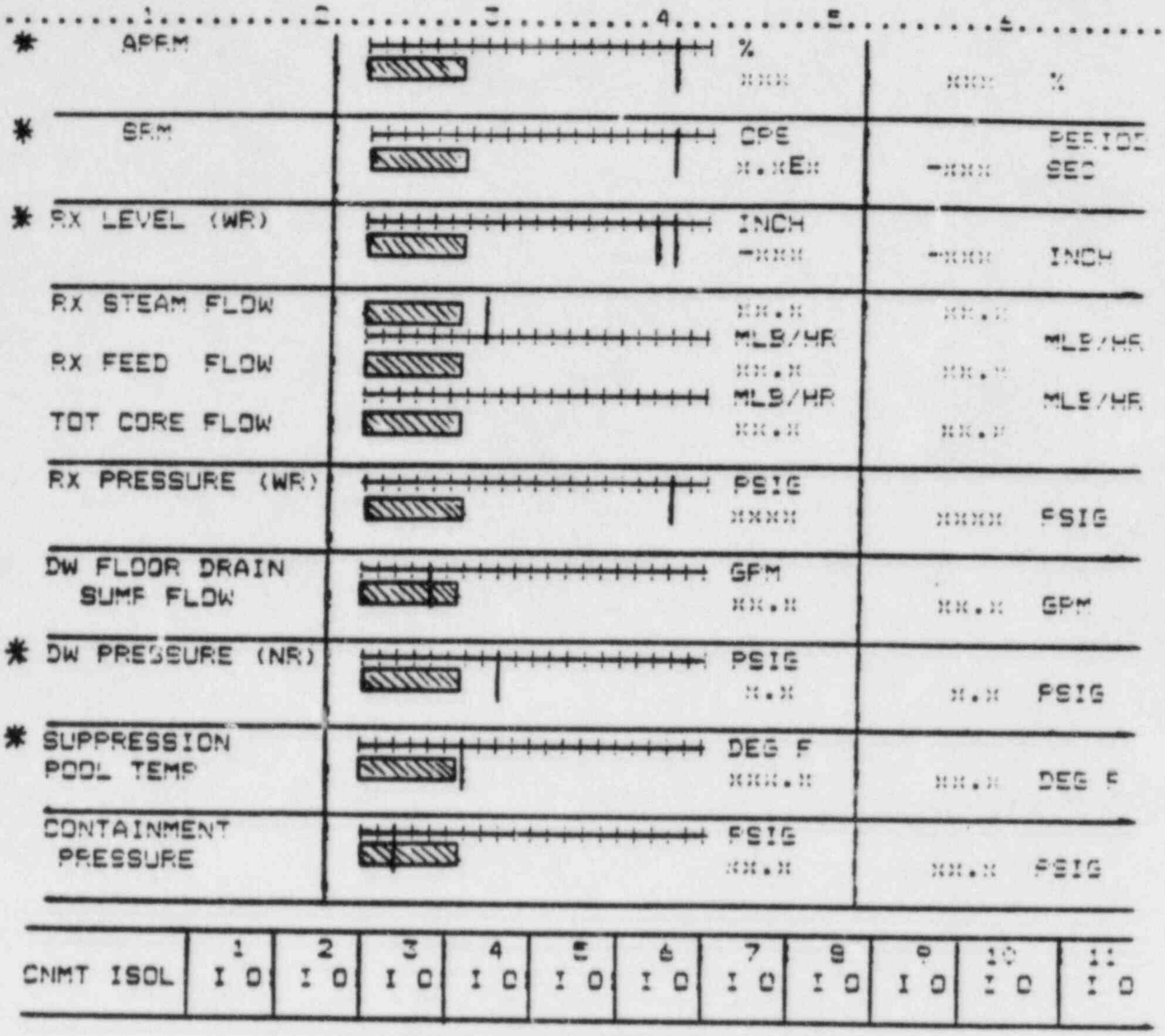
- The CPS detailed Control Room Design Review (DCRDR) is in progress. The CPS DCRDR integrates the emergency response activities of Supplement 1 to NUREG-0737.

- The human factors review of the process computer in the DCRDR will include the SPDS. This in an independent review by our consultants (Torry Pines Technology) using the methodology in NUREG-0700. The procedure used to conduct this review will be checked to ensure that the items not reviewed in the October 1983 review will be covered. Special attention will be given to the findings identified in the preliminary review of October 1983.

Schedule

- The SPDS human factors review to be completed early 1985.

- The CPS DCRDR is scheduled to be completed in June 1985.



SPDS SS DISPLAY

* RX WTR LEVEL -xxxx.x IN	SUPP POOL LVL xx.x FT
DW PRESS xx.x PSIG	SUPP POOL TEMP xxx.x F
DW TEMP xxx.x F	CNMT PRESS xx.x PSIG
SRV STATUS (OPEN/CLOSED)	CNMT TEMP xxx.x F
DW FL SUMP FLOW xx.x GPM	CNMT H2 CONC xxx.x %
SDV A LEVEL xx GAL	SDV B LEVEL xx GAL

ALARM INITIATED DISPLAY

Enclosure 4

5.0 SECONDARY CONTAINMENT/RADIOACTIVITY RELEASE CONTROL GUIDELINE

Purpose

The purpose of this guideline is to:

a) Protect equipment in the secondary containment.

b) Limit radioactivity release to the secondary containment.

c) Maintain secondary containment integrity.

d) Limit radioactivity release outside of the primary and secondary containments.

Entry Conditions

Section SC, Secondary Containment Control, should be entered if any of the following conditions are reached:

a) Secondary containment differential pressure at or above 0 inches of water.

b) Any secondary containment area temperature at or above alarm setpoint.

c) Any secondary containment HVAC cooler differential temperature at or above alarm setpoint.

d) Fuel Building Exhaust Vent Plenum radiation level at or above alarm setpoint.

e) Any secondary containment area radiation level at or above its alarm setpoint.

f) Any secondary containment floor drain sump water level at or above high high alarm setpoint.

Section RR, Radioactivity Release Control, should be entered if offsite radioactivity release rate requires an Alert.

Operator Actions

SC Secondary Containment Control

SC-1 Verify all appropriate automatic actions have occurred and manually perform any that have not:

a. VF isolation

Caution #27

b. SGTS initiation
c. VF supply fan trip

Caution #24

SC-2 IF At any time VF isolates

AND

SGTS cannot be started

AND

VF exhaust radiation level is below the isolation setpoint

THEN Restart the VF system

SC-3 Operate available area coolers and available secondary containment HVAC

SC-4 **IF** Any area temperature is at or above its alarm point

OR

Any radiation level exceeds its alarm point

OR

Any floor drain sump level cannot be restored and maintained below its alarm point

THEN Isolate all systems discharging into the area except:

a. systems required to shutdown the reactor
b. systems required to assure adequate core cooling
c. systems required to suppress a working fire

AND

Establish or verify that Secondary Containment has been established.

SC-5 **IF** A primary system is discharging into an area

THEN Before any area temperature, any area radiation level, or area water level reaches its maximum safe operating level:

a. Place the Mode switch in SHUTDOWN.
b. Perform Reactor Scram off normal procedure concurrently with this procedure.
c. Proceed to cold shutdown. Perform COOLDOWN RC/CD concurrently with this procedure.

SC-6

IF A primary system is discharging into an area

AND

either:

a. Area temperature exceeds its maximum safe operating level in more than one area

OR

b. Area radiation level exceeds its maximum safe operating level in more than one area

OR

c. Area water level exceeds its maximum safe operating level in more than one area

THEN Emergency RPV Depressurization is required. Enter Contingency #2, EMERGENCY RPV DEPRESSURIZATION, and execute it concurrently with this procedure.

RR Radioactivity Release Control

RR-1 Isolate all primary systems that are discharging into areas outside the primary and secondary containments except:

a. systems required to assure adequate core cooling

b. systems required to shutdown the reactor

RR-2 IF Offsite radioactivity release rate approaches or exceeds the release rate which requires a General Emergency.

AND

A primary system is discharging outside the primary and secondary containment

THEN Emergency RPV Depressurization is required. Enter Contingency #2, EMERGENCY RPV DEPRESSURIZATION, and execute it concurrently with this procedure.