VERMONT YANKEE
NUCLEAR POWER CORPORATION

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REPLY TO:
ENGINEERING OFFICE
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FRAMINGHAM, MASSACHUSETTS 01701
TELEPHONE 617-872-8100

2.C.2.1
FVY 83-30

April 19, 1983

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Office of Nuclear Reactor Regulation
Mr. Domenic B. Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing

References: (a) License No. DPR-28 (Docket No. 50-271)
(b) Letter, USNRC to All Operating Licenses, Generic Letter 82-33, dated December 17, 1982

Subject: NUREG-0737, Supplement I - Proposed Integrated Plan for Emergency Response Capability

Dear Sir:

Reference (b) requested that we provide you, pursuant to 10CFR50.54(f), with an intended schedule and a description of our proposed plans with regard to emergency response capability at our facility. A summary of our integrated plan, responding to the criteria detailed in Reference (b), is provided in the enclosures to this letter. Because of the unique inter-relationship between our proposed resolution to each of the subject items, it should be noted that any change in the schedule or scope of any one item will, in all likelihood, impact the scope and schedule of our overall program.

As you will note in the enclosures, certain long-range schedular details and integrated plan activities are yet to be finalized. We propose to work closely with our NRC project manager in finalizing the specific detailed schedule for each of the NUREG-0737, Supplement I, items. We believe that

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ADD: W. Paulson

Recd 4/21/83

this approach is the only way to insure that the final program addresses the criteria of NUREG-0737, Supplement I, while taking into account the resources available and our plant unique characteristics.

Very truly yours,

VERMONT YANKEE NUCLEAR POWER CORPORATION

L. H. Heider
Vice President

LHH/dd

COMMONWEALTH OF MASSACHUSETTS)
)ss
MIDDLESEX COUNTY)

Then personally appeared before me, L. H. Heider, who, being duly sworn, did state that he is a Vice President of Vermont Yankee Nuclear Power Corporation, that he is duly authorized to execute and file the foregoing request in the name and on the behalf of Vermont Yankee Nuclear Power Corporation and that the statements therein are true to the best of his knowledge and belief.

J. B. Sinclair
My Commission Expires

Notary Public
June 1, 1984

DETAILED CONTROL ROOM DESIGN REVIEW

I. NRC IMPLEMENTATION CRITERIA

A. Conduct a Control Room Design Review to identify human engineering discrepancies. The review shall consist of:

1. The establishment of a qualified multi-disciplinary review team program incorporating accepted human engineering principles.

2. The use of function and Task Analysis (that had been used as the basis for developing emergency operating procedures, technical guidelines, and plant-specific emergency operating procedures) to identify Control Room operator tasks and information and control requirements during emergency operation. This analysis has multiple purposes and should also serve as the basis for developing training and staffing needs and verifying SPDS parameters.

3. A comparison of the display and control requirements with a Control Room inventory to identify missing displays and controls.

4. A Control Room survey to identify deviations from accepted human factors principles. This survey will include, among other things, an assessment of Control Room layout, the usefulness of audible and visual alarm systems, the information recording and recall capability, and the Control Room environment.

B. Assess which human engineering discrepancies are significant and should be corrected. Select design improvements that will correct those discrepancies. Improvements that can be accomplished with an enhancement program (paint-tape-label) should be done promptly.

C. Verify that each selected design improvement will provide the necessary correction, and can be introduced in the Control Room without creating any unacceptable human engineering discrepancies because of significant contribution to increased risk, unreviewed safety questions, or situations in which a temporary reduction in safety should occur. Improvements that are introduced should be coordinated with changes resulting from other improvement programs such as SPDS, operator training, new instrumentation (Regulatory Guide 1.97, Revision 2), and upgraded emergency operating procedures.

II. NRC DOCUMENTATION CRITERIA

A. All licensees shall submit a program plan within two months of the start of the Control Room review that describes how Items I.A.1, I.A.2, and I.A.3 will be accomplished.

B. All licensees shall submit a summary report of the completed review outlining proposed Control Room changes, including their proposed schedules for implementation. The report will also provide a summary justification for human engineering discrepancies with safety significance to be left uncorrected or partially corrected.

III. RESPONSE

Item I.A, I.B, and I.C

A Control Room Design Review has been performed at Vermont Yankee. This review was conducted in accordance with the review criteria developed by the BWR Owners Group, except as discussed below.* This criteria has been submitted to you via Reference (a). The review was conducted using a multi-disciplinary review team consisting of a human factors consultant, instrument and control engineers, systems engineers, operators, and a project manager. Accepted human factors engineering principles were incorporated into our review program, as demonstrated by our use of operator interview forms and hardware checklists.

The Control Room Design Review included the use of task analyses to identify operator tasks as well as information and control requirements during emergency operation. Operational aspects of the Control Room design were evaluated through the use of task analyses and walkthroughs of function-oriented, preliminary, Emergency Operating Procedures for reactivity control, pressure control, vessel inventory, containment inventory, and containment heat removal. This effort provided us with a comparison of the display and control requirements of the procedures to the existing control board instrumentation. Additional displays or controls which would enhance operation were identified at that time. However, since these procedures were in a preliminary form, we intend to revalidate this part of the Control Room Design Review following finalization of the draft Emergency Operating Procedures.

Following the revalidation effort discussed above, we will develop a schedule for implementing any necessary Control Room enhancements. Minor enhancements (paint-tape-label) will be made promptly, with major enhancements (additional displays or controls), if found to be necessary, being made over the subsequent two refueling outages (1985 and 1986). Verification that each of the selected enhancements can be introduced into the Control Room without creating any unacceptable human engineering discrepancies will be done coincident with the assessment of which discrepancies are significant and warrant correction. The schedule for completing the Control Room enhancements assumes no unforeseen difficulties with procurement and installation of necessary equipment.

*PWR Owners Group criteria called for the use of Emergency Procedure Guidelines to identify Control Room Operator tasks and information and control requirements during emergency operation. We used function-oriented preliminary Emergency Operating Procedures for this task.

Our Control Room Design Review also included a Control Room survey to identify deviations from accepted human factors principles. The results of this effort will be included in our Control Room Design Review Report.

Items II.A and II.B

We do not intend to submit a Control Room Design Review program plan. As stated above, we have already completed a major portion of the review using the BWR Owners Group criteria as a guideline. That criteria, submitted to you via Reference (a), services as our program plan. However, we do intend to submit a summary of our final Control Room Design Review Report to you approximately 60 days following the startup from the 1986 outage.

REGULATORY GUIDE 1.97

APPLICATION TO EMERGENCY RESPONSE FACILITIES

I. NRC IMPLEMENTATION CRITERIA

Regulatory Guide 1.97 provides data to assist Control Room operators in preventing and mitigating the consequences of reactor accidents.

A. Control Room

Provide measurements and indication of Type A, B, C, D, and E variables listed in Regulatory Guide 1.97 (Revision 2). Individual licensees may take exceptions based on plant-specific design features. BWR incore thermocouples and continuous off-site dose monitors are not required pending their further development and consideration as requirements. It is acceptable to rely on currently installed equipment if it will measure over the range indicated in Regulatory Guide 1.97 (Revision 2), even if the equipment is presently not environmentally qualified. Eventually, all the equipment required to monitor the course of an accident would be environmentally qualified in accordance with the pending Commission rule on environmental qualification.

Provide reliable indication of the meteorological variables (wind direction, wind speed, and atmospheric stability) specified in Regulatory Guide 1.97 (Revision 2) for site meteorology. No changes in existing meteorological monitoring systems are necessary if they have historically provided reliable indication of these variables that are representative of meteorological conditions in the vicinity (up to about 10 miles) of the plant site. Information on meteorological conditions for the region in which the site is located shall be available via communication with the National Weather Service. these requirements supersede the clarification of NUREG-0737, Item III.A.2.2.

B. Technical Support Center (TSC)

The Type A, B, C, D, and E variables that are essential for performance of TSC functions shall be available in the TSC.

1. BWR incore thermocouples and continuous off-site dose monitors are not required pending their further development and consideration as requirements.

2. The indicators and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements.

C. Emergency Operations Facility (EOF)

1. Those primary indicators needed to monitor containment conditions and releases of radioactivity from the plant shall be available in the EOF.

2. The EOF data indications and associated circuitry shall be of reliable design but need not meet Class 1E, single-failure or seismic qualification requirements.

II. NRC DOCUMENTATION CRITERIA

A. The licensee shall submit a report describing how it meets these requirements. The submittal should include documentation which may be in the form of a table that includes the following information for each Type A, B, C, D, and E variable shown in Regulatory Guide 1.97 (Revision 2).

1. instrument range

2. environmental qualification (as stipulated in guide or state criteria)

3. seismic qualification (as stipulated in guide or state criteria)

4. quality assurance (as stipulated in guide or state criteria)

5. redundance and sensor(s) location(s)

6. power supply (e.g., Class 1E, non-Class 1E, battery backed)

7. location of display (e.g., Control Room board, SPDS, chemical laboratory)

8. schedule (for installation or upgrade)

Deviations from the guidance in Regulatory Guide 1.97 (Revision 2) should be explicitly shown, and supporting justification or alternatives should be presented.

III. RESPONSE

Items I.A, I.B, I.C, and II.A

We are in the process of performing an engineering assessment of the Type A, B, C, D, and E variables listed in Regulatory Guide 1.97, Revision 2. This assessment is currently scheduled to be complete by June 1, 1984. This assessment will include:

o An evaluation of the necessary measurements and indications for the Control Room;

- Justification for any deviations from the guidance criteria of Regulatory Guide 1.97, Revision 2; and

- A complete listing of the plant parameters to be displayed in our Technical Support Center (TSC) and Emergency Operations Facility (EOF). These lists will supercede the lists previously submitted to you as Tables I, II, and III of Reference (b).

Following the completion of this assessment, we will establish a schedule for the installation of any necessary modifications. It is our intent to submit the Regulatory Guide 1.97 Final Report, including the schedule for installation of any necessary modifications, by August 1, 1984.

UPGRADE EMERGENCY OPERATING PROCEDURES (EOPs)

I. NRC IMPLEMENTATION CRITERIA

The use of human factored, function oriented, emergency operating procedures will improve human reliability and the ability to mitigate the consequences of a broad range of initiating events and subsequent multiple failures or operator errors, without the need to diagnose specific events.

A. In accordance with NUREG-0737, Item I.C.1, reanalyze transients and accidents and prepare Technical Guidelines. These analyses will identify operator tasks and information and control needs. The analyses also serve as the basis for integrating upgraded emergency operating procedures and the Control Room design review and verifying the SPDS design.

B. Upgrade EOPs to be consistent with Technical Guidelines and an appropriate procedure Writer's Guide.

C. Provide appropriate training of operating personnel on the use of upgraded EOPs prior to implementation of the EOPs.

D. Implement upgraded EOPs.

II. NRC DOCUMENTATION CRITERIA

A. Submit Technical Guidelines to NRC for review. NRC will perform a pre-implementation review of the Technical Guidelines. Within two months of receipt of the Technical Guidelines, NRC will advise the licensees of their acceptability.

B. Each licensee shall submit to NRC a procedures generation package at least three months prior to the date it plans to begin formal operator training on the upgraded procedures. NRC approval of the submittal is not necessary prior to upgrading and implementing the EOPs. The procedures generation package shall include:

1. Plant-Specific Technical Guidelines -- plant-specific guidelines for plants not using generic technical guidelines. For plants using generic technical guidelines, a description of the planned method for developing plant-specific EOPs from the generic guidelines, including plant-specific information.

2. A Writer's Guide that details the specific methods to be used by the licensee in preparing EOPs based on the Technical Guidelines.

3. A description of the program for validation of EOPs.

4. A brief description of the training program for the upgraded EOPs.

III. RESPONSE

Items I.A and I.B

Reference (c) documents the NRC's acceptance of the BWR Owners Group Emergency Operating Procedures (EOPs) Technical Guidelines. As an active participant in the development of these Technical Guidelines, we will use this information as the basis for the preparation of plant-specific, symptom based EOPs, which are currently being developed.

The upgraded EOPs will be consistent with the intent of the Technical Guidelines as well as our procedure Writers Guide.

Items I.C and I.D

Our current EOP upgrade program calls for the extensive training of operating personnel prior to implementation of the EOPs. Training of the six operations shifts will take approximately four months and is scheduled to begin bu October 1, 1984. Implementation of the EOPs is expected by February 1, 1985.

Item II.A

As discussed in our response to Items I.A and I.B, the Technical Guidelines developed by the BWR Owners Group have been approved by the NRC [see Reference (c)].

Item II.B

It is our present intent to submit our Procedures Generation Package by July 1, 1984, which is approximately three months prior to the date we plan to begin formal operating personnel training. This package will include:

- A description of the method used for developing plant-specific EOPs from the Generic Technical Guidelines;
- A Writer's Guide that details the specific methods used in preparing our EOPs;
- A description of our program validation of the new EOPs; and
- A summary description of our training program for the EOPs.

EMERGENCY RESPONSE FACILITIES

I. NRC IMPLEMENTATION CRITERIA

A. Technical Support Center (TSC)

The TSC is the on-site technical support center for emergency response. When activated, the TSC is staffed by pre-designated technical, engineering, senior management, and other licensee personnel, and five pre-designated NRC personnel. During periods of activation, the TSC will operate uninterrupted to provide plant management and technical support to plant operations personnel, and to relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations. Teh TSC will perform EOF functions for the Alert Emergency Class and for the Site Area Emergency Class and General Emergency Class until the EOF is functional.

1. The TSC will be:

   a. Located within the site protected area so as to facilitate necessary interaction with Control Room, OSC, EOF, and other personnel involved with the emergency.

   b. Sufficient to accommodate and support NRC and licensee pre-designated personnel, equipment and documentation in the center.

   c. Structurally built in accordance with the Uniform Building Code.

   d. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

   e. Provide with radiological protection and monitoring equipment necessary to assure that radiation exposure to any person working in the TSC would not exceed 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

   f. Provided with reliable voice and data communications with the Control Room and EOF and reliable voice communications with the OSC, NRC Operations Centers and state and local operations centers.

   g. Capable of reliable data collection, storage, analysis, display and communication sufficient to determine site and

regional status, determine changes in status, forecast status and take appropriate actions. The following variables shall be available in the TSC:

(1) The variables in the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Revision 2) that are essential for performance of TSC functions; and

(2) The meteorological variables in Regulatory Guide 1.97 (Revision 2) for site vicinity and National Weather Service data available by voice communication for the region in which the plant is located.

Principally those data must be available that would enable evaluating incident sequence, determining mitigating actions, evaluating damages and determining plant status during recovery operations.

h. Provided with accurate, complete and current plant records (drawings, schematic diagrams, etc.) essential for evaluation of the plant under accident conditions.

i. Staffed by sufficient technical, engineering, and senior designated licensee officials to provide needed support, and be fully operational within approximately 1 hour after activation.

j. Designed taking into account good human factors engineering principles.

B. Operations Support Center (OSC)

When activated, the OSC will be the on-site area separate from the Control Room where pre-designated operations support personnel will assemble, A pre-designated licensee official shall be responsible for coordinating and assigning the personnel to tasks designated by Control Room, TSC, and EOF personnel.

1. The OSC will be:

a. Located on-site to serve as an assembly point for support personnel and to facilitate performance of support functions and tasks.

b. Capable of reliable voice communications with the Control Room, TSC, and EOF.

C. Emergency Operations Facility (EOF)

The EOF is a licensee controlled and operated facility. The EOF provides for management of overall licensee emergency response, coordination of radiological and environmental assessment,

development of recommendations for public protection actions, and coordination of emergency response activities with Federal, State, and local agencies.

When the EOF is activated, it will be staffed by pre-designated emergency personnel identified in the emergency plan. A designated senior licensee official will manage licensee activities in the EOF.

Facilities shall be provided in the EOF for the acquisition, display, and evaluation of radiological and meteorological data and containment conditions necessary to determine protective measures. These facilities will be used to evaluate the magnitude and effects of actual or potential radio-active releases from the plant and to determine dose projections.

1. The EOF will be:

   a. Located and provided with radiation protection features as described in Table 1 (previous guidance approved by the Commission) and with appropriate radiological monitoring systems.

   b. Sufficient to accommodate and support Federal, State, local, and licensee pre-designated personnel, equipment, and documentation in the EOF.

   c. Structurally built in accordance with the Uniform Building Code.

   d. Environmentally controlled to provide room air temperature, humidity and cleanliness appropriate for personnel and equipment.

   e. Provided with reliable voice and data communications facilities to the TSC and Control Room, and reliable voice communication facilities to OSC and to NRC, State, and local emergency operations centers.

   f. Capable of reliable data collection, storage, analysis, display and communication of information on containment conditions, radiological releases, and meteorology sufficient to determine site and regional status, determine changes in status, forecast status and take appropriate actions. Variables from the following categories that are essential to EOF functions shall be available in the EOF:

      (1) Variables from the appropriate Table 1 or 2 of Regulatory Guide 1.97 (Revision 2); and

      (2) Meteorological variables in Regulatory Guide 1.97 (Revision 2) for site vicinity and regional data available via communication from the National Weather Service.

g. Provided with up-to-date plant records (drawings schematic diagrams, etc.), procedures, emergency plans and environmental information (such as geophysical data) needed to perform EOF functions.

h. Staffed using Table 2 (previous guidance approved by the Commission) as a goal. Reasonable exceptions to goals for the number of additional staff personnel and response times for their arrival should be justified and will be considered by NRC staff.

i. Provided with industrial security when it is activated to exclude unauthorized personnel and when it is idle to maintain its readiness.

j. Designed taking into account good human factors engineering principles.

III. RESPONSE

Following the declaration of an emergency, the activities of the emergency organization are coordinated in a number of dedicated emergency response facilities. The emergency response organization is activated in part, or wholly, dependent upon the operator's recognition of Emergency Action Levels which represent a particular emergency condition. In the initial phases of an emergency, the Control Room coordinates all aspects of emergency response and corrective action required to restore safe conditions. As additional response personnel report, certain emergency response functions are transferred to personnel located in the Technical Support Center, Operations Support Center, and the Emergency Operations Facility.

Item I.A

A Technical Support Center (TSC) has been established to direct post-accident evaluations and assist in recovery operations of the plant. The TSC is located on the second floor of the Administration Building in close proximity to the Control Room and will be radiologically habitable to the same degree as the Control Room for postulated accident conditions.

The TSC contains radiological protection and monitoring equipment to assess radiological habitability. The TSC was part of the original plant construction and built in accordance with the uniform building code.

The location and communications equipment of the TSC facilitate interaction and coordination with the Control Room, EOF, Operations Support Center (OSC), NRC, and State emergency facilities. The determination of how the TSC will ultimately be provided access to information on critical plant parameters will follow the completion of our Regulatory Guide 1.97 assessment and subsequent development of our computer based display system.

The TSC will accommodate sufficient technical staff needed to evaluate the plant condition. Communications with the Yankee Engineering Support Center in Framingham, Massachusetts, will reduce the necessity for a large number of additional technical support personnel at the TSC. The Yankee Nuclear Services Division will dispatch technical support personnel as necessary to support the TSC in diagnosing accident conditions. Also, the TSC has access to as-built drawings of plant systems and equipment, system flow diagrams, cable/wiring diagrams, technical manuals, emergency plans, plant procedures, and current plant records which are essential for evaluation of, and recommendations for, the plant under accident conditions.

Item I.B

The Operations Support Center (OSC) is located on the first floor of the Administration Building. The OSC provides a general assembly area for chemistry and health physics technicians and auxiliary operators who would be utilized to effect corrective actions in support of the emergency situation. A predesignated OSC Coordinator directs the center activities. Communications between the OSC, Control Room, TSC, and EOF are provided via in-plant telephone extensions and a page/intercom system.

Item I.C

The primary Emergency Operations Facility (EOF) is located about .35 miles from the plant, in the Governor Hunt House on Governor Hunt Road in Vernon, Vermont. The overall responsibility for the functions performed in the EOF belongs to the Site Recovery Manager. The Site Recovery Manager acts as the principle spokesman of Vermont Yankee concerning all issues where an interface with off-site governmental authorities is required. The EOF Coordinator is responsible for continued evaluation of all licensee activities related to an emergency having, or potentially having, environmental radiological consequences. The Governor Hunt House has sufficient assembly space to accommodate the emergency organization and responding representatives from government and industry responsible for evaluating off-site consequences. The EOF provides information needed by off-site authorities for implementation of off-site Emergency Plans in addition to a centralized meeting location for key representatives from the agencies.

The EOF maintains communications capability with all other emergency centers. The primary EOF has access to meteorological data for site vicinity via voice communications with the plant Control Room, and the region via voice communications with the Yankee Engineering Support Center in Framingham, Massachusetts, on request.

The determination of how the EOF will ultimately be provided access to plant status, radiological and environmental data necessary for the development of recommendations for public protective actions, will follow the completion of our Regulatory Guide 1.97 assessment and subsequent development of our computer based display system.

Federal and State and licensee pre-designated personnel who will be accommodated in the primary EOF will utilize materials available within

the EOF to evaluate the magnitude and effects of actual or potential radioactive releases from the plant and to project doses. The primary EOF is provided with communication links to the TSC, Control Room, OSC, NRC, State emergency facilities, and the Media Center.

The primary EOF contains radiation monitoring equipment and procedures to assess EOF radiological habitability. This radiological surveillance provides early warning of any need to transfer the critical EOF functions of dose projection and protective action recommendation to the alternate EOF.

An alternate EOF has been established at the Vermont Yankee Corporate Office located in Brattleboro, Vermont, approximately 8.75 miles from the site. Although this is slightly less than the ten-mile distance indicated in Table 1 of Supplement I to NUREG-0737, we find this facility acceptable. We reach this conclusion on the basis of Vermont Yankee power level (504 MWe net vs. 1000 MWe for which the ten-mile plume EPZ determination was made) and the fact that this facility is housed in a large brick and concrete structure.

This EOF facility use plan, makes the issue of near site EOF protection factors and ventilation isolation with HEPA filters moot. Under Design Bases Accident Conditions, the 30-day whole body dose at the near site EOF is less than 5 rems (neglecting protection factors and assuming 100% occupancy). This is below the EPA Protective Action Guideline of 5 rem suggested as a level for evacuation of the public and far below the 25 rem limit for emergency work PAGs as identified in NUREG-0654 (Planning Standard K). The conclusion of our evaluation of the near site EOF against the NRC's guidelines is that with our approach additional modifications, such as upgrading the facility to increase protection factors and adding HEPA filtered ventilation systems, are not necessary because we satisfy the functional intent of the guidelines.

Each EOF can be provided with industrial security when activated to exclude unauthorized personnel and when idle to maintain its readiness.

This description of our Emergency Response Facilities discussed above supercedes the information previously submitted to you via Reference (d). However, more details on the integration and layouts of these facilities can be found in the Vermont Yankee Emergency Plan.

ENCLOSURE 5

SAFETY PARAMETER DISPLAY SYSTEM (SPDS)

I. NRC IMPLEMENTATION CRITERIA

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. Although the SPDS will be operated during normal operations as well as during abnormal conditions, the principal purpose and function of the SPDS is to aid the Control Room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phase of an accident.

A. Each operating reactor shall be provided with a Safety Parameter Display System that is located convenient to the Control Room operators. 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.

B. The Control Room instrumentation required (see General Design Criteria 13 and 19 of Appendix A to 10CFR50) provides the operators with the information necessary for safe reactor operation under normal, transient, and accident conditions. The SPDS is used in addition to the basic components and serves to aid and augment these components. Thus, requirements applicable to control room instrumentation are not needed for this augmentation (e.g., GDC 2, 3, and 4 in Appendix A; 10CFR Part 100; single-failure requirements). The SPDS need not meet requirements of the single-failure criteria, and it need not be qualified to meet Class 1E requirements. The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems. The SPDS need not be seismically qualified, and additional seismically qualified indication is not required for the sole purpose of being a backup for SPDS. 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. Furthermore, operators should be trained to respond to accident conditions both with and without the SPDS available.

C. There is a wide range of useful information that can be provided by various systems. This information is reflected in such staff documents as NUREG-0696, NUREG-0835, and Regulatory Guide 1.97.

Prompt implementation of an SPDS can provide an important contribution to plant safety. The selection of specific information

that should be provided for a particular plant shall be based on engineering judgement of individual plant licensees, taking into account the importance of prompt implementation.

D. This SPDS display shall be designed to incorporate accepted human factors principles so that the displayed information can be readily perceived and comprehended by SPDS users.

E. 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. radioactive control

5. containment conditions

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

F. Integration

Prompt implementation of an SPDS is a design goal and of primary important. The schedule for implementating SPDS should not be impacted by schedules for the Control Room design review and development of function-oriented emergency operating procedures. For this reason, licensees should develop and propose an integrated schedule for implementation in which the SPDS design is an input to the other initiatives. If reasonable, this schedule will be accepted by NRC.

II. NRC DOCUMENTATION CRITERIA

A. The licensee shall prepare a written safety analysis describing 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 included symptoms of severe accidents. Such analysis, along with the specific implementation plan for SPDS, shall be reviewed as described below.

B. The licensee's proposed implementation of an SPDS system shall be reviewed in accordance with the licensee's technical specifications to determined whether the changes involve an unreviewed safety question or change of technical specifications. If they do, they shall be processed in the normal fashion with prior NRC review. If the changes do not involve an unreviewed safety question or a change in the technical specifications, the licensee may implement such change without prior approval by NRC or may request a pre-implementation review and approval. If the changes are to be

implemented without prior NRC approval, the licensee's analysis shall be submitted to NRC promptly on completion of review by the licensee's off-site safety review committee. Based on the results of the NRC review, the Director of IE or the Director of NRR may request or direct the licensee to cease implementation if a serious safety question is posed by the licensee's proposed system, or if the licensee's analysis is seriously inadequate.

III. RESPONSE

Items I.A through I.F

It is our intent to display the plant safety parameters in an area of the Control Room convenient to the operators. This information will be continuously displayed in a manner such that the plant status can be readily assessed. The method chosen for display will be suitably isolated from any electrical or electronic interference with equipment and sensors that are in use for safety systems.

The determination of what parameters will be displayed will be predicated on the results of our Control Room Design Review, Regulatory 1.97 assessment, and finalization of draft Emergency Operating Procedures. Completion of these activities is essential in determining the makeup of the display system. In addition, accepted human factors principles will be considered during the detailed design of the display system.

At a minimum, information will be provided relative to:

- reactivity control,
- reactor core cooling and heat removal from the primary system,
- reactor coolant system integrity,
- radioactive control, and
- containment conditions.

Item II.A

Following our determination of what information will be included as part of our display system, we will prepare a written safety analysis which describes the basis for the determination. The safety analysis will include an assessment that the displayed information is sufficient to assess the safety status of each identified function for a wide range of events, including symptoms of severe accidents. The safety analysis will be submitted following the review by our off-site safety review committee.

ENCLOSURE 6

REFERENCES

(a) Letter, BWROG to USNRC, BWROG - 8151, dated August 25, 1981

(b) Letter, VYNPC to USNRC, FVY 82-10, dated February 5, 1982

(c) Letter, USNRC to all BWRs; Safety Evaluation of "Emergency Procedure Guidelines, Revision 2", NEDO-24934, June 1982; dated February 8, 1982

(d) Letter, VYNPC to USNRC, WVY 80-07, dated January 8, 1980