UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 2 S 1978

MEMORANDUM FOR: Raymond F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

FROM: Edson G. Case, Deputy Director
Office of Nuclear Reactor Regulation

SUBJECT: STATUS OF ACRS RECOMMENDATIONS

Your memorandum of August 28, 1978, requested additional clarification regarding a number of recommendations made by the Committee during the period January 1, 1977 through September 30, 1977. These were initially forwarded by your memorandum of December 1, 1977, to which we responded on April 21, 1978.

The enclosure provides the additional clarification requested by your August 28, 1978, memorandum.

Edson G. Case

Edson G. Case, Deputy Director
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

cc: H. Denton
S. Levine
C. Smith
R. Minogue
J. Davis
R. Boyd
R. DeYoung
R. Mattson
V. Stello
W. Russell
W. Minners
D. Bunch

APPENDIX A

201st Meeting, January 6-8, 1977

1. Question:

The Committee looks forward to receiving the Staff evaluation of the feedwater monitoring program. Feedwater piping vibrations will be discussed at a future meeting of the Fluid Hydraulics Subcommittee.

Response:

The staff has not yet completed its review of the feedwater monitoring program at Beaver Valley 1. However, the following is a status of our evaluation to date.

The feedwater system hydraulic transient monitoring program at the Beaver Valley Power Station, Unit 1 was carried out during the period of March 1 to August 1, 1977. The purpose of this program was to verify the adequacy of system modifications to prevent certain feedwater line vibrations and to obtain data, in the event of such vibrations, that would facilitate an assessment of piping stresses and allow an analysis of causative factors. The licensee reported the results of its monitoring program by letter dated November 23, 1977, and in response to staff questions, it submitted additional information by letter dated March 17, 1978.

The staff has reviewed the information submitted by the licensee and has not found a recurrence of the same type of vibrations that occurred once in November, once in December 1976 and once in January 1977 during the startup of Beaver Valley, Unit 1. The data showed only one indication of possible vibration that occurred on July 17, 1977, approximately 10 seconds after a load rejection test from 50% power to zero power. The above cited three incidents of significant pipe vibration occurred during operation between 30% to 50% power. The licensee has stated that the indications of vibration on July 17, 1977, were due to the anomalous response of certain instruments to mechanical vibrations caused by closure of the feedwater pump check valves and the turbine stop valves. The staff has not reached a conclusion on this point. The staff intends to request additional information and complete its evaluation in February 1979.

However, since modifying the internals of the feedwater control valves in February 1977, the Beaver Valley, Unit 1 facility has not experienced a recurrence of the same type of vibration that occurred prior to the modification. This is a positive indication that the cause of those vibrations has been eliminated.

APPENDIX A

201st Meeting, January 6-8, 1977

3. Question:

The resolution is acceptable for North Anna. It is not clear how the North Anna resolution is to be translated into a generic resolution. Clarification is requested.

Response:

The resolution for North Anna on a postulated fuel handling accident inside containment is being pursued by the staff in a generic manner. Briefly, the staff is requiring that a plant possess the capability for prompt detection of any radioactivity release inside containment and automatic containment isolation using redundant radiation monitors. The staff is revising its Standard Review Plan (Section 15.7.4) to reflect this consideration.

The revised acceptance criteria for consideration of fuel handling accidents inside containment are as follows (extracted from revised Section 15.7.4):

> Where an applicant proposes that fuel handling operations inside containment occur only when containment is isolated, or where the containment is continuously vented to the environment via an iodine filter system, this is acceptable. Where fuel handling operations inside containment occur when the containment is open to the environment (i.e., with a containment purge exhaust system) the proposed design is acceptable if it possesses the capability for prompt detection and automatic containment isolation by use of redundant radiation monitors.

The revised review procedures (extracted from Section 15.7.4) for consideration of this matter are as follows:

> The proposed systems intended to mitigate the consequences of a fuel handling accident inside containment are reviewed. Where an applicant proposes that fuel handling will occur only when the containment is isolated, this is acceptable and no radiological consequences need be calculated. Where fuel handling

operations occur only when the containment is exhausted to the environment via an ESF filter system, this is acceptable and the radiological consequences should be calculated giving appropriate credit for this system. Where the containment will be open during fuel handling operations (as with a containment purge exhaust system), the reviewer should verify that a prompt detection and automatic containment isolation capability is provided and that an independent evaluation of the consequences shows that the resulting doses are within the acceptance criteria given in Section II.2. A review should be made of the applicant's analysis and should include examination of the type, location and redundancy of the radiation monitors intended to detect an activity release within containment and verification that detection is followed by automatic containment isolation. The reviewer should assess the time required to isolate the containment. This should include the instrument line sampling time (where appropriate), detector response time and containment purge isolation valve actuation and closure time. The containment is considered isolated only when the purge isolation valves are fully closed and seated. The applicant's analysis should be reviewed regarding the travel time of any activity release starting from its release point above the refueling cavity or transfer canal and including travel time in ducts or ventilation systems until it reaches the inner containment purge isolation valve. Where the applicant claims credit for dilution or mixing of a release due to natural or forced convection inside containment prior to release, this is reviewed and assessed. Refs. 3 and 4 may be consulted and used by the reviewer for guidance in estimating dilution and mixing. The time required for the release to reach the inner isolation valve is compared to the time required to isolate containment. If the time required for the release to reach the isolation valve is longer than the time required to isolate containment, then essentially no release outside of containment occurs, and the reviewer's assessment will reflect this. If the time required for the release to reach the isolation valve is less than that required to isolate containment, and no mixing or dilution credit can be given, the reviewer should assume that the entire activity release escapes from the containment in evaluating the consequences. Where mixing and dilution within containment isolation, the radiological consequences will be reduced compared to the entire activity release by the degree of mixing and dilution occurring prior to containment isolation.

APPENDIX A

202nd Meeting, Februar, 10-12, 1977

1. Question:

   The document referenced by Dr. Moeller should have been proposed ANSI Standard N13/42, "Performance Specifications for Reactor Emergency Radiological Monitoring Instrumentation", (Tables 1, 2 and 3).

   Response:

   The staff has noted this item. The reference appears to be to BNWL-1635, dated May 1972 which is incorporated as Reference 3 in Reg. Guide 1.97, Rev. 1.

APPENDIX A

202nd Meeting, February 10-12, 1977

3. Question:

   It is not clear from the response what final action the NRC Staff has taken. Clarification is requested.

   Response:

   The staff erred in its previous response. We should have noted that although the regulations exclude consideration of nuclear weapons detonation from consideration, electromagnetic interference (EMI) that might arise from other sources is being investigated.

   NUREG-0153 discusses the effect of EMP from a nuclear weapon and two ORNL technical reports on the subject. Other external sources of electromagnetic interference, such as lightning, would produce amplitudes that would be only a small fraction of that from a nuclear weapon. Thus the following quote from NUREG-0153 would apply to these other possible sources.

   > "In all nuclear power plants, the reactor and some of the protection system circuitry are located within the containment building, which is either built of steel plate or is a concrete structure lined with steel plate. In both cases, the shielding from EMP provided by the steel plate is excellent and there should be no adverse effects within the containment structure. However, a substantial part of the protection system circuitry is outside the containment, in the control room, the cable spreading room, and in portions of the auxiliary building, where essential auxiliary systems are located.
   >
   > The control room and auxiliary buildings are normally constructed of reinforced concrete of heavy construction since they are built to withstand tornado missiles, differential pressures and seismic events. The multiple courses of reinforcing bars in the walls and ceilings of these structures should provide substantial attenuation of EMP. It appears that up to 30 to 40 db of attenuation are available from this sort of heavily reinforced concrete construction. Further shielding is

provided by steel cabinets, cable raceways, and electrical conduits for wire and cable runs inside these structures.

The ORNL reports find that the most serious effects would be on digital logic circuits. They find that analog-type control circuits are more resistant to pulse damage. There is also a strong effect from large pulses on solid state circuitry, because the solid state elements (diodes, transistors, etc.) are typically unable to accept large temporary overloads as are vacuum tube elements. Digital computers with solid state components are probably the most vulnerable kind of equipment to EMP exposures."

Effects of inplant EMI phenomena from sources other than nuclear weapons, especially on safety-related digital equipment, are being addressed on recently submitted standard design applications; e.g., BSAR-205 (RPS-II), CESSAR (CPC) and RESAR-414 (IPS).

Typical of our approach on the standard designs is the recently completed review of ANO-2. The staff expressed concern with the susceptibility of the ANO-2 core protection calculator (CPC) system to EMI. A position was developed and a test program was set up to verify that the proper operation of the CPC system will not be compromised by radiated or conducted noise signals that can be expected during nuclear power plant operation. The test procedure and test results are addressed in the ANO-2 SER. The susceptibility tests for EMI radiation and conduction were run in accordance with MIL requirements.* Also as a guide, the staff utilizes in its review RDT Standard CI-IT, "Instrumentation and Control Equipment Grounding and Shielding Practices", as a methodology to minimize the effects of EMI phenomena. A common practice within industry is to provide a shield around a twisted pair of wires and ground one end of the shield. This minimizes the capacitive coupling from the external voltage sources to the pair of wires inside the shield. In conjunction with shielding, in-line filters are used to suppress the undesirable

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* 1. MIL-STD-416A; Military Standard Electromagnetic Interference Requirements for Equipment.

2. MIL-STD-464; Military Standard, Electromagnetic Interference Characteristics, Measurement of.

frequencies. In order to evaluate the effectiveness of the shielding and filtering, it is necessary to measure the actual levels of frequencies of EMI in-situ and evaluate their impact on equipment/system susceptibility. At the present time, such tests are being conducted at the ANO-2 plant. Similar rev[illegible]s are being conducted on the PDA applications listed above.

We believe that the ongoing review of th[illegible]f EMI phenomena on safety-related digital equipment will ide[illegible]need and priority for any further study on this subject for [illegible]rious safety-related features of nuclear power plants.

APPENDIX A

204th Meeting, April 7-9, 1977

1a. Question:

The response relative to pump flywheels is acceptable for the present, but there is no indication of the schedule for ultimate resolution of this matter. The Committee recommends that the Staff make quarterly reports until a technological solution to this problem is identified.

Response:

The staff now considers this issue to be resolved. The requirement for the maximum acceptable flaw size in pump flywheel material in Regulatory Guide 1.14 is primarily based on the capabilities of current manufacturing processes and inspection methods, since the calculated critical flaw size that could result in failure is larger by a significant margin. The specified maximum acceptable flaw size is well above the detectable limits of current inspection methods. The specified flaw size is also within the capabilities of current manufacturing processes and available from commercial sources. The staff believes that sufficient technical basis exists to support our current requirements. However, the staff remains open to receive and review any new information that might support a change to our position.

APPENDIX A

204th Meeting, April 7-9, 1977

1b. Question:

The Committee would like to be kept informed regarding the development and application of this probabilistic methodology to this subject.

Response:

Subsequent to the July 1, 1977 staff report to Libarkin from Denton, NRR prepared a research request on probabilistic flood assessments. Enclosed is a copy of the October 26, 1977 memo to the Office of Nulcear Regulatory Research suggesting activities in this area. The intent of the proposed research is to assess and potentially improve the acceptability of methodology associated with estimating probabilities of severe floods. No formal program has yet been received from research on this subject.

OCT 26 1977

MEMORANDUM FOR: Saul Levine, Director
Office of Nuclear Regulatory Research

FROM: Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

SUBJECT: RESEARCH REQUIREMENTS FOR EVALUATION OF MARGINS AVAILABLE IN FLOOD PROTECTION OF NUCLEAR POWER PLANTS (RR-NRR 77-16)

NRR requests RES to initiate confirmatory research related to evaluating the margins inherent in flood protection of nuclear power plants. Both WASH-1400 and licensing experience indicate that identification of such margins is important to either confirming that present practice is adequate, or for modifying future practice. The contacts for this work are W. S. Bivins and L. G. Hulman, both at 492-7288.

BACKGROUND

Our current methods and criteria for analysis of flood potential and for flood protection are summarized in the following documents:

a. R.G. 1.59, Design Basis Floods for Nuclear Power Plants;

b. American National Standard N-170, Standards for Determining Design Basis Flooding at Power Reactor Sites.

c. R.G. 1.70, Section 2.4, Standard Format and Content of SARs for Nuclear Power Plants;

d. R.G. 1.102, Flood Protection for Nuclear Power Plants.

The assumption and underlying practice in this subject area is that a nuclear power plant hardened against the most severe flooding conditions reasonably probable is adequate to protect the public health and safety. Potential flooding conditions are analyzed deterministically using techniques and procedures evolved from practice by other Federal agencies (primarily the Corps of Engineers, NOAA, FPC, and Bureau of Reclamation). Furthermore, these techniques and procedures consider the range of causative mechanisms, including tropical storms, large and small scale extra tropical precipitation and wind storms, geoseismic activity and dam failures. No assessment is made of the probability of the flood conditions postulated. Futhermore, no evaluation is made of the likelihood of failure of flood protection, the consequences of failure, the residual risks inherent in inadequate flood condition/flood protection criteria, or the degree of conservatism associated with the present methodology.

We have followed the evolution of probabilistic techniques with considerable interest, particularly those associated with WASH-1400, and have attempted to utilize their application in flooding assessments on several occasions. Our latest application is summarized in a memo to ACRS from H. R. Denton, dated July 1, 1977 (copy enclosed) which indicates our concerns related to probabilistic techniques applied to estimating the likelihood of severe floods. Our primary concerns include the following:

a. A single measure of an event outcome, such as water level or discharge, is generally used as an indicator of event magnitude. No differentiation is made as to the cause of the event, however, and experience indicates that a flood record contains events caused by at least two completely different phenomena (e.g., tropical and extra tropical storms). A typical flood record may not contain a large enough sample of floods caused by each type of event to be representative. Furthermore, even if a flood record is not considered composed of mixed events, the representativeness of a relatively short-term record for prediction of very low likelihood events may be questionable.

b. The selection of confidence limits that (1) minimize the residual error in estimates of event magnitudes, and correspondingly, (2) minimize the range of event likelihood.

c. If likelihood estimates are made using dependent and independent components of event magnitude (e.g., rainfall magnitude, areal distribution of rainfall, ground wetness, etc.), how are individual component confidence limits reconciled to minimize the residual error in estimates of the outcome magnitude and outcome likelihood?

Flood protection requirements vary considerably from site to site. For example, if all safety-related facilities are located above design basis flood levels, no flood protection provisions are required. Many sites fall in this category; others do not. Prior to the issuance of Reg. Guide 1.102, flood protection provisions at those sites susceptible to flooding often included many provisions requiring emergency action to provide external water barriers. With the advent of Reg. Guide 1.102, hardened protection has been the staff goal such that water barriers are permanently in place. Based upon this history, designs and costs of providing flood protection vary considerably from site to site.

To assess the overall risk, we have consistently concluded that a plant accommodating a design basis flood condition (which could be caused by a severe precipitation, dam failure, hurricane, wave action, or seismically induced event) is adequate. No detailed assessment has been made of the overall risk of a severe flood for which either flood protection is inadequate, or for the likelihood and consequences of a failure of design flood protection. Both of these situations should be assessed to assure that (1) flood protection requirements are adequate, and (2) residual risks are appropriately minimized.

INFORMATION NEEDS

Our information needs are divided into two categories: assessments of methodology uncertainties for applying probabilistic methods to predicting severe flooding events, and the residual risk associated with present flood protection requirements. We are aware of no programs in any of the National Laboratories that are compatible with the work proposed herein. There are, however, several researchers that have evaluated extreme natural phenomena probabilistically, and residual risk assessments in the area of earthquakes have been undertaken. Futhermore, a numerical evaluation of accident risk is under study with PNL, and may provide a basis for the work requested herein.

Specifically, the following material should be provided:

a. assess the long term representativeness of stream, lake and coastal flood records with particular emphasis on causative mechanisms (including hurricanes, large scale extra tropical storms and thundershowers).

b. identify acceptable methodology (or methodologies) for selecting confidence limits that (1) minimize residual risks in extreme event magnitude evaluation at design levels of 10^{-6} to 10^{-7} per year, and (2) minimize the uncertainty in probability estimates at the same design levels.

c. if individual components of flood events are used to assess event likelihood, instead of a single outcome, identify an acceptable methodology (or methodologies) that also satisfies b. above.

d. assess the likelihood of flood protection not performing its required function and the resulting potential consequences.

We recognize, however, that a conclusion from this research may be that extreme flood events cannot be predicted with an acceptable level of confidence.

The desired time frame for completing this activity is the first quarter of FY 79 to allow for development of any indicated changes in staff review methodology and changes in Standard Review Plans.

LICENSING IMPACT

This program may provide a basis for a probabilistic assessment of the flood potential at nuclear power plants. Overall, identification of the safety margin available in flood potential and flood protection will provide a basis for considerations of revisions to standard review plans and safety guides presently employing deterministic approaches.

RESEARCH EFFORT

No assessment of the level of effort required has been made. We suggest proposals be sought and we will be glad to participate in their review. This request has been discussed informally with Ian Wall and Jerry Harbour of your office.

VALUE/IMPACT ASSESSMENT

No quantification of flood likelihood is available for use to judge the utility of regulatory requirements. A well considered probabilistic analysis will provide the basis for (1) maintaining the present level of flood evaluation and protection requirements, (2) requiring less protection, or (3) requiring more protection, and changing present evaluation methodologies.

Three alternatives to this proposal were considered as follows:

a. continue the present methodology;

b. arbitrarily increase or decrease the level required for flood protection by simply adding or subtracting an increment of elevation; or

c. requiring flood protection redundancy.

The latter two alternatives are considered purely arbitrary without the results of the requested research. The first alternative, business as usual, will be continued until we can evaluate the results of the requested research based upon not only our own experience that no historical flood event has produced conditions worse than postulated, but similar experience of other Federal agencies.

Original Signed By
E. G. Case

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosures:
As Stated

cc: w/enclosure
H. Denton
D. Muller
L. Rubenstein
F. Miraglia
D. Wigginton
DSE ADs
DSE BCs
J. Harbour, RES
L. Beratan, OSD
J. Knight
R. Tedesco
I. Sihweil
V. Benaroya
W. Bivins
HES Personnel
C. Jupiter, RES

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DSE:ST:HMB*	DSE:ST*	DSE	NRR	NRR
LGHulman:jd	WPGammill	HRDenton	LRubenstein	EGCase
10/16/77	10/17/77	10/ /77	10/ /77	/ /77

APPENDIX A

205th Meeting, May 5-6, 1977

1. Question:

The Staff reply is not responsive to the inquiry that addressed the procedures for identifying drawings and descriptive material that should be withheld from public disclosure. Apparently, the decision to withhold has been turned over to the Commission. For the present, information is being withheld on a proprietary basis. The Committee is interested in the ground rules for establishing what information should be withheld assuming that the proprietary alternative can be implemented. The memorandum from Goller suggests that the licensee will make judgments concerning information to be safeguarded. It is not clear whether the NRC Staff has a basis for testing the licensees' judgment. Clarification is requested.

Response:

Generally, the staff will withhold from public disclosure any drawing or descriptive material which details, displays, identifies or amplifies a licensee's or applicant's site specific method for safeguarding licensed special nuclear material or security measures taken for the physical protection of a licensed facility or plant in which licensed special nuclear material is processed or used. This position is believed to be fully justified pursuant to the provisions of 10 CFR 2.790(d)(1). It is recognized that in some cases, however, it may be necessary or prudent for the staff to disclose information which would not significantly or adversely affect a licensee's or applicant's physical security system. Such disclosure, though, would normally only concern generic physical security requirements or matters of common knowledge.

In other cases, the staff can and has challenged the validity of a licensee's or applicant's request to withhold information from public disclosure. Should the staff challenge such a request, however, they must determine:

1. Whether the information has been held in confidence by its owner.

2. Whether the information is of a type customarily held in confidence by its owner and whether there is a rational basis therefore.

3. Whether the information was transmitted to and received by the Commission in confidence.

4. Whether the information is available in public sources.

5. Whether public disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of the owner of the information.

Should a request for withholding pursuant to the above be denied, the Commission notifies the licensee and provides the licensee with a statement of reasons for the denial.

In any instance, a balancing of the interests of the person or agency urging nondisclosure and the public interest in disclosure is made by the staff.

APPENDIX A

205th Meeting, May 5-6, 1977

2. Question:

Formation of a damage control team should receive consideration. A fairly modest investment might lead to a considerable improvement in response time and capability. An ACRS Subcommittee will follow up this response with a meeting to discuss the subject.

Response:

It appears at this time that a specific "damage control team" is not needed to provide an effective response to acts of sabotage. This position is based on the assumption that the Safeguards Contingency Plans, which the licensees are required to submit to the Commission in accordance with 10 CFR 50.34(d) and Appendix C to 10 CFR Part 73, will provide licensee response forces with predetermined measures or actions to be initiated should a sabotage event be attempted.

The main purpose of the contingency plans is to identify credible events capable of disrupting plant operations, e.g., attempted sabotage. The plans require statements of the objective(s) to be achieved for each event and the actions to be accomplished by the response force.

In addition to the above, the licensee is required to provide the Commission with an Emergency Plan in accordance with the provisions of Appendix E to 10 CFR Part 50. The plan must provide reasonable assurance that appropriate measures can and will be taken in the event of an emergency to protect the public health and safety.

In view of the above, it is currently believed that the requirements detailed in each plan, including the duties defined and assigned to specific personnel, provide the same degree of control as that of a specially designated "damage control team".

APPENDIX A

206th Meeting, June 9-10, 1977

2. Question:

See response to 205th Meeting, item 2.

Response:

We assume this statement implies a Committee interest in the establishment of "damage control teams" to handle fires. Such a team is, in fact, in existence at the Zion Station.

The staff requires that licensees establish "fire brigades" for immediate response to fire threats. A general description of the staff requirements as regards minimum manning levels for the fire brigades is attached as Enclosure 1. For the Zion Station, these manpower requirements are established in the Technical Specifications. Copies of applicable pages of these Technical Specifications are included as Enclosure 2.

The staff has evaluated the overall fire protection program for the Zion Station and has reported the results of its review in a Safety Evaluation. This Safety Evaluation is incorporated as an enclosure to a letter of March 10, 1978, to the licensee, which issued amendments regarding fire protection to the Zion Station units. Copies of this letter with the Safety Evaluation were previously forwarded to the Committee.

The staff considers that its evaluation of the Zion Station for fire protection, as reported in the Safety Evaluation, includes the matters raised by R. Pollard in his testimony on North Anna before the ASLB.

.0 ADM[illegible]STRATIVE CONTROLS

.1 Organization, Review, Investigation and Audit

A. The Station Superintendent shall have overall full-time responsibility for safe operation of the facility. During periods when the Station Superintendent is unavaiable, he shall designate this responsibility to an established alternate who satifies the ANSI N18.1 experience requirements for plant manager.

B. The corporate management which relates to the operation of this station is shown in figure 6.1.1.

C. The normal functional organization for operation of the station shall be as shown in Figure 6.1.2. The shift manning for the station shall be as shown in Figure 6.1.3. A Fire Brigade of at least 5 members shall be maintained on-site at all times. The fire brigade shall not include the minimum shift crew necessary for safe shutdown of the plant (4 members) or any personnel required for other essential functions during a fire emergency.

D. Qualifications of the station management and operating staff shall meet minimum acceptable levels as described in ANSI "Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971 with the exception of the Radiological Chemical Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8 September, 1975. The individual filling the position of Administrative Assistant shall meet the minimum acceptable level for "Technical Manager" as described in 11.2.4 of

E. Retraining and replacement training of Station personnel shall be in accordance with ANSI N18.1,"Selection and Training of Nuclear Power Plant Personnel", dated March 8, 1971. A training program for the Fire Brigade shall be maintained under the direction of the Station Fire Marshall and shall meet or exceed the requirements of Section 27 of the NFPA Code - 1975, except that Fire Brigade training will be conducted quarterly.

F. Retraining shall be conducted at intervals not exceeding two years.

G. The Review and Investigative Function and the Audit Function of activities affecting quality during facility operations shall be constituted and have the responsibilities and authorities outlined below:

The Supervisor of the Offsite Review and Investigative Function shall be appoined by the Vice President of Construction Production, Licensing and Environmental Affairs. The Audit Function shall be the responsibility of the Manager of Quality Assurance and shall be independent of operations.

a. <u>Offsite Review and Investigative Function</u>

The Supervisor of the Offsite Review and Investigative function shall: (1) provide directions for the review and investigative function and appoint a senior participant to provide appropriate direction, (ll) select each participant for this function, (lll) select a complement of more than one participant who collectively possess backround and qualifications in the subject matter under review to provide

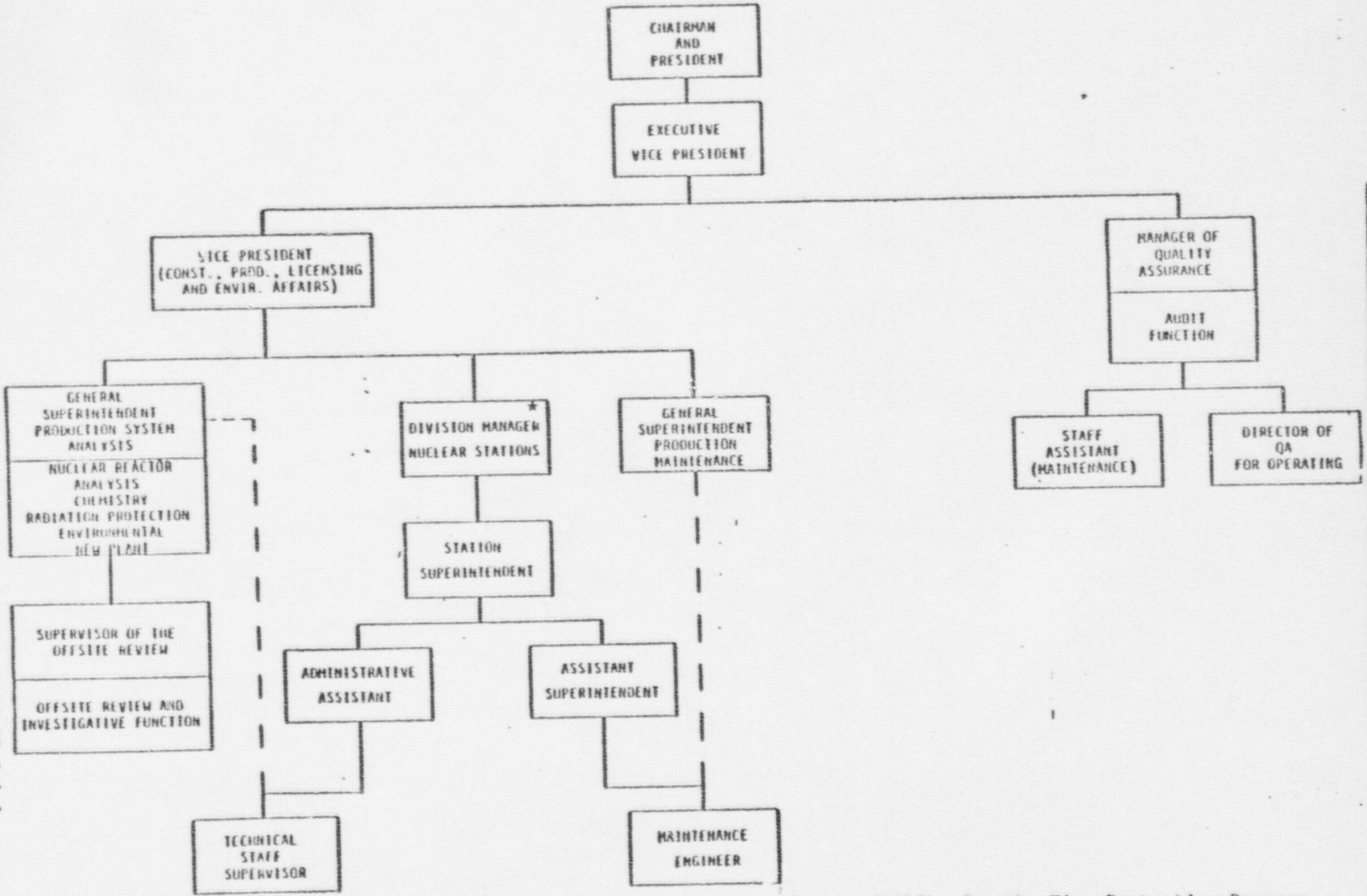
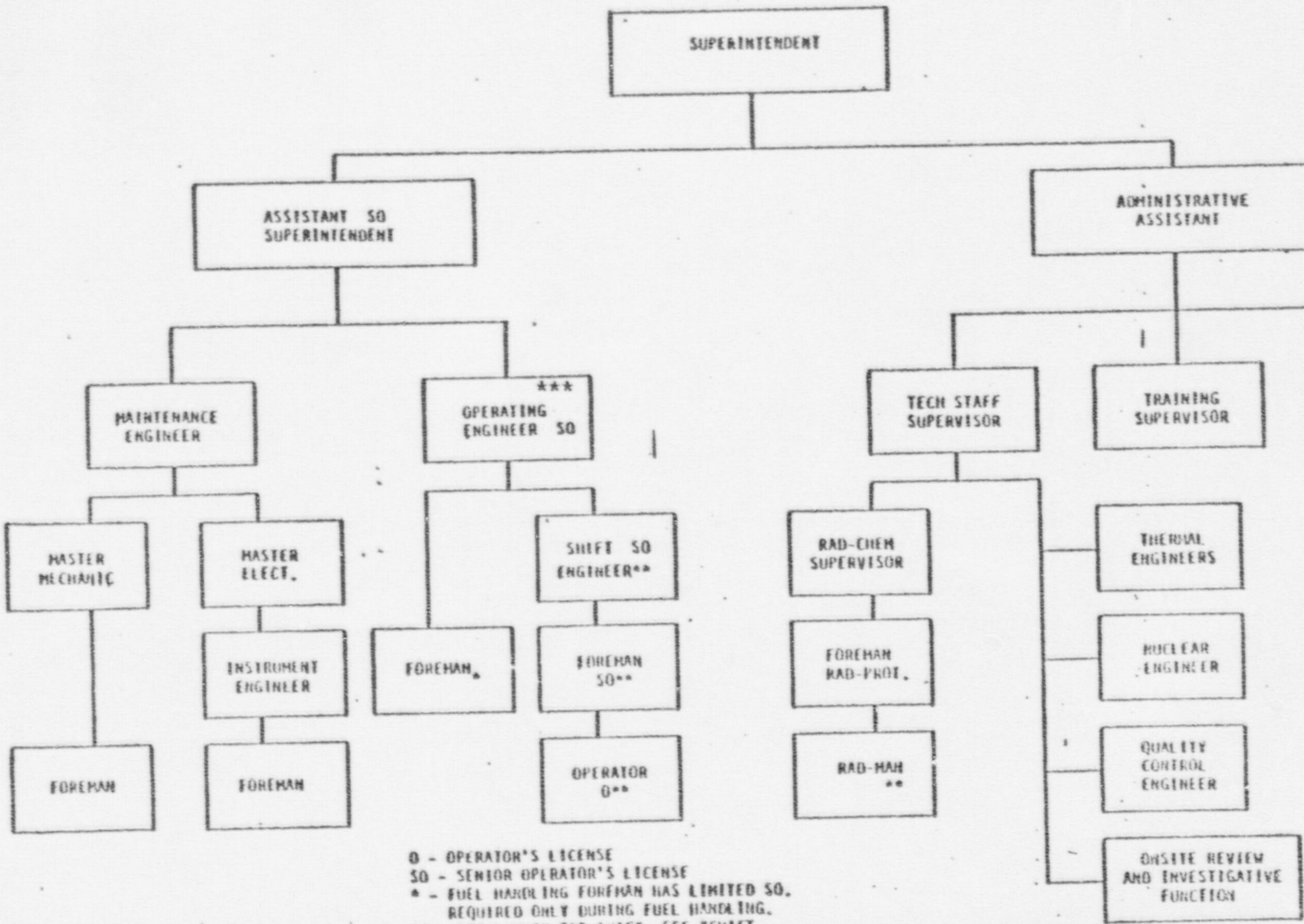


FIGURE 6.1-1
Corporate Organization

*Responsibility for the Fire Protection Program

Amendment No. ~~28~~ 36 & 33



O - OPERATOR'S LICENSE
SO - SENIOR OPERATOR'S LICENSE
* - FUEL HANDLING FOREMAN HAS LIMITED SO. REQUIRED ONLY DURING FUEL HANDLING.
** - FOR NUMBER PER SHIFT, SEE "SHIFT MANNING CHART", FIGURE 6.1-3
*** - Responsible for implementation of Fire Protection Program

Zion Station Organization FIGURE 6.1-2

MANPOWER REQUIREMENTS FOR OPERATING REACTORS

The NRC has established requirements for personnel at operating reactors for purposes of plant operation, industrial security, and fire fighting. The following discussion considers the extent to which plant personnel assigned to either plant operation or security may also be temporarily allowed to man a fire brigade in the event of a fire for a single unit facility and sets forth an acceptable sharing scheme for operating reactors.

Summary of Manpower Requirements

1. Fire Brigade: The staff has concluded that the minimum size of the fire brigade shift should be five persons unless a specific site evaluation has been completed and some other number justified. The five-man team would consist of one leader and four fire fighters and would be expected to provide defense against the fire for an initial 30-minute period. See Attachment A for the basis for the need for a five-man fire brigade.

2. Plant Operation: Standard Review Plan Section 13.1.2 requires that for a station having one licensed unit, each shift crew should have at least three persons at all times, plus two additional persons when the unit is operating. For ease of reference, Attachment B contains a copy of this SRP.

3. Plant Security: The requirements for a guard force are outlined in 10 CFR Part 73.55. In the course of the staff's review of proposed security plans, a required minimum security response force will be established for each specific site. In addition to the response team, two additional members of the security force will be required to continuously man the Central Alarm Station (CAS) and Secondary Alarm Station (SAS). It is expected that many facilities will have a security organization with greater numbers of personnel than the minimum number assumed for purposes of discussion in this paper.

The NRC staff has given consideration to the appropriateness of permitting a limited degree of sharing to satisfy the requirements of plant operation, security and fire protection and has concluded that, (1) subject to certain site and plant specific conditions, the fire brigade staffing could generally be provided through operations and security personnel, and (2) the requirements for operators and the security force should remain uncompromised. Until a site specific review is completed, the following indicates the interim distribution and justification for these dual assignments, and therefore our interim minimum requirements for a typical presently operating commercial single unit facility. The staff believes that manpower for the fire brigade for multi-unit facilities is not now a problem because of the larger numbers of people generally present at the sites. Situations which do pose problems will be reviewed on a case-by-case basis.

1. Plant Operation: The staff has concluded that for most events at a single unit nuclear facility, a minimum of three operators should be available to place the reactor in a safe condition. The two additional operators required to be available at the nuclear facility are generally required to be present to perform routine jobs which can be interrupted to accomodate unusual situations that may arise. That is, there is the potential for the remaining two members of the operating crew to assume other short-term duties such as fire fighting. In light of the original rationale for providing extra plant operators to cope with off-normal conditions, it appears justified to rely on these personnel for this function. The staff recommends that one of the two operators assigned to the fire brigade should be designated as leader of the fire brigade in view of his background in plant operations and overall familiarity with the plant. In this regard, the shift supervisor should not be the fire brigade leader because his presence is necessary elsewhere if fires occur in certain critical areas of the plant.

2. Plant Security: In the event of a fire, a contingency plan and procedures will be used in deploying the security organization to assure that an appropriate level of physical protection is maintained during the event. The staff has determined that it is possible in the planning for site response to a fire, to assign a maximum of three members of the security organization to serve on the fire brigade and still provide an acceptable level of physical protection. While certain security posts must be manned continuously (e.g., CAS, SAS), the personnel in other assignments, including the response force, could be temporarily (i.e., 30 minutes) assigned to the fire brigade. In judging the merits of this allowance the underlying question is whether the minimum security force strength must be maintained continuously in the event of a plant emergency such as a fire. Further examination of this issue leads to two potential rationales for reaching an affirmative decision. First, could there be a causal connection between a fire and the security threat? Second, are there compelling policy reasons to postulate a simultaneous threat and fire?

   The first potential rationale would only be credible if, (1) the insider (posed as part of the threat definition) was an active participant in an assault and started a fire coincident with the attack on the plant or, (2) a diversionary fire was started by an attack force somewhere external to the plant itself where no equipment required for safe shutdown is located. The role of the insider will be discussed first. While 73.55 assigns an active status to the insider, the rule also requires that measures be implemented to contain his activities and thereby reduce his effectiveness. At present, these measures include background checks on plant employees, limited access to vital plant areas, badging systems and the two-man rule. Here, limited access means that only designated employees are allowed in vital areas and that their entry is controlled by either conventional locks or card-key systems. Also, if separate trains of safety equipment are involved, then either compartmentalization or the two-man rule is required. These measures to contain the insider are presently being implemented and will provide assurance that people of questionable reliability would not be able to gain employee status at a nuclear plant and should they become an employee with unescorted access, significant restraints would be interposed on the ability of such a person to carry out extensive damage to plant vital areas. Recognizing that additional safeguards may still be appropriate, the staff has recommended to the Commission that plant personnel also be required to obtain an NRC security clearance. The staff believes that the attendant background investigation associated with a clearance, in conjunction with the other 73.55 measures, will provide a high degree of assurance that plant personnel will not attempt to take an active sabotage role. If the clearance rule is adopted the staff believes some of the measures, such as the two-man rule, designed to contain the insider can be relaxed. Thus, there does not now appear to be a reasonably credible causative relationship between a fire intentionally set by an insider and the postulated external security threat. For the case of diversionary fires set external to the plant itself, adequate security forces can still be maintained by allowing only part of the fire brigade to respond while both fire fighters and security force armed responders maintain a high degree of alertness for a possible real attack somewhere else on the plant. Thus, the effective number of armed responders required by 73.55 can be maintained for external diversionary fires.

The second potential rationale concerns whether a serious, spontaneous fire should be postulated coincident with an external security threat as a design basis. In evaluating such a requirement it is useful to consider the likelihood of occurrence of this combination of events. While it is difficult to quantify the probability of the 73.55 threat, it is generally accepted that it is small, comparable probably to other design basis type events. The probability of a fire which is spontaneous and located in or in close proximity to a vital area of the plant and is serious enough to pose a significant safety concern is also small. It would appear, therefore, that the random coincidence of these two unlikely events would be sufficiently small to not require protection against their simultaneous occurrence. In addition, it should be noted that the short time period (30 minutes) for which several members of the security force would be dedicated to the fire brigade would further reduce the likelihood of coincidence.

As neither of the two potential rationales appear to preclude the use of members of the security force in the event of a fire the staff has concluded that the short assignment of security personnel from the armed response force or other available security personnel to the fire brigade under these conditions would be acceptable.

To ensure a timely and effective response to a fire, while still preserving a flexible security response, the staff believes that the fire brigade should operate in the following manner. In the event of an internal fire, all five members of the fire brigade should be dispatched to the scene of the fire to assess the nature and seriousness of the fire. Simultaneously, the plant security force should be actively evaluating the possibility of any security threat to the plant and taking any actions which are necessary to counter that threat. For external fires, a lesser number than the five-man brigade should respond for assessment and fire fighting. As the overall plant situation becomes apparent it would be expected that the most effective distribution of manpower between plant operations, security and fire protection would be made, allowing a balanced utilization of manpower resources until offsite assistance becomes available. The manpower pool provided by the plant operations personnel and security force are adequate to respond to the occurrence of a design basis fire or a security threat equivalent to the 73.55 performance requirements. It is also recognized that other, more likely combinations of postulated fires and security threats of a lesser magnitude than the design basis, could be considered. While the probabilities of these higher likelihood events may be sufficient to warrant protecting against them in combination, the manpower requirements required to cope with each event would be similarly reduced thereby allowing adequate coverage by plant personnel.

Conclusion

The staff believes that it would be reasonable to allow a limited amount of sharing of plant personnel in satisfying the requirements of plant operation, security, and fire protection. An acceptable sharing scheme would entail reliance on two plant operators and three members of the security organization to constitute the fire brigade. Since availability of the full fire brigade would only be required for fires with potential for serious damage, actual distribution of plant personnel during a plant emergency would be governed by the exigencies of the situation. Of course, all personnel assigned to the fire brigade would have to fulfill all applicable training requirements. It should also be recognized that the diversion of personnel to the fire brigade would be of short duration and that substantial additional offsite assistance would be forthcoming in accordance with the emergency and contingency plan developed for each facility. In evaluating licensee proposals for manpower sharing due consideration will also have to be made of unique facility characteristics, such as terrain and plant lay-out, as well as the overall strengths of the licensee's fire and security plans. Minimum protection levels in either area could preclude the sharing of manpower.

Staff Position

Minimum Fire Brigade Shift Size

INTRODUCTION

Nuclear power plants depend on the response of an onsite fire brigade for defense against the effects of fire on plant safe shutdown capabilities. In some areas, actions by the fire brigade are the only means of fire suppression. In other areas, that are protected by correctly designed automatic detection and suppression systems, manual fire fighting efforts are used to extinguish: (1) fires too small to actuate the automatic system; (2) well developed fires if the automatic system fails to function; and (3) fires that are not completely controlled by the automatic system. Thus, an adequate fire brigade is essential to fulfill the defense in depth requirements which protect safe shutdown systems from the effects of fires and their related combustion by-products.

DISCUSSION

There are a number of factors that should be considered in establishing the minimum fire brigade shift size. They include:

1) plant geometry and size;
2) quantity and quality of detection and suppression systems;
3) fire fighting strategies for postulated fires;
4) fire brigade training;
5) fire brigade equipment; and
6) fire brigade supplements by plant personnel and local fire department(s).

In all plants, the majority of postulated fires are in enclosed windowless structures. In such areas, the working environment of the brigade created by the heat and smoke buildup within the enclosure, will require the use of self-contained breathing apparatus, smoke ventilation equipment, and a personnel replacement capability.

Certain functions must be performed for all fires, i.e., command brigade actions, inform plant management, fire suppression, ventilation control, provide extra equipment, and account for possible injuries. Until a site specific review can be completed, an interim minimum fire brigade size of five persons has been established. This brigade size should provide a minimum working number of personnel to deal with those postulated fires in a typical presently operating commercial nuclear power station.

If the brigade is composed of a smaller number of personnel, the fire attack may be stopped whenever new equipment is needed or a person is injured or fatigued. We note that in the career fire service, the minimum engine company manning considered to be effective for an initial attack on a fire is also five, including one officer and four team members.

It is assumed for the purposes of this position that brigade training and equipment is adequate and that a backup capability of trained individuals exist whether through plant personnel call back or from the local fire department.

POSITION

1. The minimum fire brigade shift size should be justified by an analysis of the plant specific factors stated above for the plant, after modifications are complete.

2. In the interim, the minimum fire brigade shift size shall be five persons. These persons shall be fully qualified to perform their assigned responsibility, and shall include:

   One Supervisor - This individual must have fire tactics training. He will assume all command responsibilities for fighting the fire. During plant emergencies, the brigade supervisor should not have other responsibilities that would detract from his full attention being devoted to the fire. This supervisor should not be actively engaged in the fighting of the fire. His total function should be to survey the fire area, command the brigade, and keep the upper levels of plant management informed.

   Two Hose Men - A 1.5 inch fire hose being handled within a windowless enclosure would require two trained individuals. The two team members are required to physically handle the active hose line and to protect each other while in the adverse environment of the fire.

   Two Additional Team Members - One of these individuals would be required to supply filled air cylinders to the fire fighting members of the brigade and the second to establish smoke ventilation and aid in filling the air cylinder. These two individuals would also act as the first backup to the engaged team.

4. a. Assignments of personnel meeting ANSI N18.1-1971 qualifications, Section 4.3.1 or Section 4.5.1, should be made to onsite shift operating crews numbers not less than the following:

For a station having one licensed unit, each shift crew should have at least three persons at all times, plus two additional persons when the unit is operating. For a multi-unit station, each shift crew should have at least three persons per licensed unit at all times, plus one additional person per operating unit.

b. Operator license qualifications of persons assigned to operating shift crews should be as follows:

(1) A licensed senior operator who is also a member of the station supervisory staff should be onsite at all times when at least one unit is loaded with fuel.

(2) For any station with more than one reactor containing fuel, (1) the number of licensed senior operators onsite at all times should not be less than the number of control rooms from which the fueled units are monitored, and (2) the number of licensed senior operators should not be less than the number of reactors operating.

(3) For each reactor containing fuel, there should be at least one licensed operator in the control room at all times. Shift crew compositions should be specified such that this condition can be satisfied independently of licensed senior operators assigned to shift crews to meet the criteria of (1) and (2) above.

(4) For each control room from which one or more reactors are in operation, an additional operator should be onsite and available to serve as relief operator for that control room. Shift crew compositions should be specified such that this condition can be satisfied independently of (1), (2), and (3), and for each such control room.

c. Radiation protection qualifications of at least one person on each operating shift should be as follows:

The management of each station having one or more units containing fuel should either, (1) qualify and designate at least one member of each shift operating crew to implement radiation protection procedures, including routine or special radiation surveys using portable radiation detectors, use of protective barriers and signs, use of protective clothing and breathing apparatus, performance of contamination surveys, checks on radiation monitors, and limits of exposure rates and accumulated dose, or (2) assign a health physics technician to each shift, such assignment to be in addition to those assigned to shift operating crews in accordance with (a) and (b) above.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgment on the areas to be given attention during

APPENDIX A

208th Meeting, August 11-13, 1977

2. Question:

The intent of this request was to be sure that open items identified for the PDA are addressed when the PDA is used in a construction permit application. The response is not clear on this matter; clarification is requested.

Response:

The ACRS memorandum of August 28, 1978 clarified the intent of the original ACRS request made during the 208th meeting (August 11-13, 1977). The original request was misinterpreted in the staff response of April 28, 1978. The request is now understood to be a concern "that open items identified for the PDA are addressed when the PDA is used in a construction permit application".

Open items identified during a PDA review are addressed in the review of a construction permit application referencing a PDA or PDA application.

In most cases, the open items in a PDA review are resolved prior to the issuance of a PDA; for such items the resolution of the issue is applied to any and all construction permit application(s) (or other PDA application(s)) under review which reference the PDA application under review.

In some cases, a PDA may be issued subject to the resolution of certain issues, i.e., the resolution of an issue might not be completed at the time of issuance of the PDA. An applicant referencing a PDA will be required to resolve any outstanding issue, within the applicant's scope of responsibility, as stated in the NRC Safety Evaluation Report (SER) pertaining to the original issue of the PDA referenced by the applicant. If one or more supplements to the SER have been issued, the requirements, if any, to be addressed by a referencing applicant would be identified in the supplement (or supplements) pertaining to the issuance of any and all amendments to the original PDA, during the effective period of the PDA.

For example, the application for a construction permit for the Phipps Bend nuclear facility (Docket Nos. 050-553, -554) addressed open issues then remaining on the GESSAR 238 (Nuclear Island) PDA (No. 1).

Similarly, applicants referencing the original BSAR-205 PDA (No. 12) will be required to address the matter of reactor cooldown using only safety-grade systems, as required by paragraph (6) of the PDA.

APPENDIX A

208th Meeting, August 11-13, 1977

3. Question:

According to our records we have not received a Staff response on this item.

Response:

A response to this item was inadvertently omitted from our original memorandum. The original Committee request is as follows:

> 3. Dr. Bush requested that the NRC staff discuss its requirements for snubbers, the potential consequences from snubber failures, and methods for assuring that they will in fact work when needed.
>
> The staff in its reports on snubbers at the August 11-13, 1977 ACRS meeting addressed all of the ACRS concerns. Further staff actions are presented in Task Action Plan A-13, "Snubbers". The staff believes that the completion of this task action plan will resolve all outstanding concerns of the ACRS.

APPENDIX A

209th Meeting, September 8-10, 1977

3. Question:

It is not clear whether the premises described by Portland G.E. with respect to failure to isolate containment is representative of the case in question. The intent of the inquiry was to obtain an assessment concerning the habitability of a control room with degraded containment capability subsequent to an accident where radiation releases are a variable. For example, the intent was to determine whether a large number of fuel cladding failures coincident with a LOCA and partially ineffective containment closure could influence the habitability of control rooms. One approach might be to consider the effects of 1% fuel clad perforation, 5% fuel clad perforation, and 50% fuel clad perforation as possible conditions coincident with a LOCA and incomplete containment as a way of assessing control room habitability contingencies. The Committee would appreciate a response on this matter.

Response:

Control room habitability is reviewed by the staff for the case of a postulated LOCA using the source term of Regulatory Guides 1.3 or 1.4, coupled with the operation of engineered safety features designed to mitigate the consequences of the event and assuming the containment is leaking at the design leak rate. The single failure criterion is invoked in evaluating the performance of engineered safety features designed to mitigate the consequences of this event. Thus, where iodine removal sprays are employed for example, the evaluation assumes 1 out of the 2 spray trains fails to function. The radiological consequences are required to be within the criteria given in GDC 19 of Appendix A to 10 CFR Part 50. In comparing the hypothesized source term to that which is representative for a fuel failure of 10%, for example, the staff estimates that the hypothesized source term is approximately 250 to 500 times greater than that represented by failure of 10% of the fuel rods. On this basis, the staff concludes that realistic fuel failure rates would be within the GDC 19 limits even for containment leak rates significantly higher than the design leak rate.

APPENDIX B

201st Meeting, January 6-8, 1977

2b. Question:

The Committee recommends that the Staff provide guidance and a schedule for implementation of Reg. Guide 1.97.

Response:

The staff currently is in the process of revising its approach toward implementation of Regulatory Guide 1.97, Revision 1. A description of the present status and the proposed future course of action is provided in the attached memorandum from R. H. Vollmer, dated October 12, 1978. As noted in the draft schedule, included with Mr. Vollmer's memorandum, we plan to discuss this matter further with the Committee. It now appears that we could be prepared to meet with the Committee during its January 1979 meeting.

OCT 1 2 1978

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director for Light Water Reactors, DPM
Robert L. Tedesco, Assistant Director for Plant Systems, DSS
Brian K. Grimes, Assistant Director for Engineering and Projects, DOR
Darrell G. Eisenhut, Assistant Director for Systems and Projects, DOR
Frank Schroeder, Acting Assistant Director for Reactor Safety, DSS

FROM: Richard H. Vollmer, Assistant Director for Site Analysis, DSE

SUBJECT: IMPLEMENTATION OF R. G. 1.97

As you know we have been in the process of implementing R.G. 1.97, Rev. 1 for some time within the program defined by TAP A-34. This program initially envisioned the use of the lead plant concept to work out the details of the sometimes complex requirements of the regulatory guide. Our experience with this approach to date has not been fruitful although in the process, guidance has been developed upon which further efforts in implementation can be based.

Because of the difficulties encountered in utilizing the lead plant concept, we are abandoning this approach and propose to proceed in a more straight-forward conventional manner as discussed in the attached outline. Please review this proposed approach and the enclosed draft schedule for implementation and provide your comments to me by November 1, 1978.

/s/ R. W. Houston

for Richard H. Vollmer, Assistant Director
for Site Analysis
Division of Site Safety and
Environmental Analysis

Enclosures:
As stated

cc: R. DeYoung
R. Mattson
V. Stello
S. Varga
L. Crocker

Distribution
Central File
AAB Reading
AAB File (TAP-A-34)
G. Chipman
R. Vollmer

OFFICE	AAB:DSE	AAB:DSE	AD:SA:DSE		
SURNAME	GChipman/bm	RWHouston	RHVollmer		
DATE	10/12/78	10/12/78	10/12/78		

Proposed Course of Action for Implementation of R.G. 1.97 10/12/78

Task Action Plan A-34 was developed to provide a systematic approach to implementation of R.G. 1.97. It called for the use of the lead plant concept in working out the detailed requirements in two operating phases; one phase for position C-3 of the guide (instrumentation for beyond design basis events) and the other for implementation of position C-1 (instrumentation for design basis events). Our work to date in dealing with the selected lead plants has been unsatisfactory for developing guidance. As a result, we believe that implementation should proceed without further reliance on the lead plant concept. TAP A-34 is being revised to reflect a different approach as described below.

All plants will be required to implement position C-3 (except for C-3.d) on a reasonable time schedule. C-3.d is excluded because no current instruments are available which will fulfill the requirements needed to monitor the large and variable releases for identifiable release points (C-3.d). A contract to determine the feasibility of and overall performance requirements for such instrumentation will be let. The results of this contract will be utilized to provide appropriate criteria for implementation and backfit of position C-3.d.

Beginning with the review of the New Haven application, we anticipate requiring applicants to provide the analysis required in position C-1. Our evaluation of these analyses will determine the specific instrumentation needs related to design basis events. As these instrumentation needs are identified on current and future licensing reviews, a determination will be made concerning backfit of such instrumentation on operating plants.

Draft Schedule for Implementation of R. G. 1.97

Revise TAP to reflect proposed approach.	11/78
Present approach to ACRS.	12/78
Approve revised TAP.	12/78
Implement position C-1 review on Haven application.	12/78
Letter to applicants and licensees on all LWR plants requiring backfit of position C-3 (except C-3.d).	1/15/79
Let contract for feasibility study of instruments required by position C-2.d and develop design criteria.	1/15/79
Required response date by applicants - committment and schedule for SAR submittal and installation.	3/15/79
Complete development of design criteria for position C-3.d instruments.	6/15/79
Letter to applicants and licensees on all LWRs requiring implementation of position C-3.d in accordance with enclosed guidance.	8/1/79
Completion date for position C-3.d implemented on all plants - to be developed.	?

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OCT 25 1978

MEMORANDUM FOR: Domenic B. Vassallo, Assistant Director
for Light Water Reactors, DPM
Robert L. Tedesco, Assistant Director
for Plant Systems, DSS
Brian K. Grimes, Assistant Director
for Engineering and Projects, DOR
Darrell G. Eisenhut, Assistant Director
for Systems and Projects, DOR
Frank Schroeder, Acting Assistant Director
for Site Analysis, DSS

FROM: Richard H. Vollmer, Assistant Directro
for Site Analysis

SUBJECT IMPLEMENTATION OF R. G. 1.97

My October 12, 1978 memo included an enclosure that described a proposed course of action for implementation of Regulatory Guide 1.97, Revision 1. The proposal incorrectly recommended that we require applicants to provide the analysis required in Position C.1 beginning with the review of the Haven application. In fact, we propose that the analysis be required beginning with the New Haven application. Please make your comments based on this correction by November 1, 1978.

John T. Collins
Richard H. Vollmer, Assistant Director
for Site Analysis
Division of Site Safety and
Environmental Analysis

cc: R. DeYoung
R. Mattson
V. Stello
S. Varga
✓L. Crocker

APPENDIX B

201st Meeting, January 6-8, 1977

2d. Question:

The Committee wishes to be kept informed regarding the results of the Licensee's reliability study, the Staff's evaluation of it, the final fix required and its generic implications, if any.

Response:

No further information is available at this time. The license requires submittal of an analysis and, if required, installation of the final fix at first refueling which is scheduled for February 1980. When the information is available we will inform the committee as requested.

APPENDIX B

201st Meeting, January 6-8, 1977

3a. Question:

The Committee desires information regarding stress levels for various structures and components required for safe shutdown and long-term cooling presented in such a manner that the margin against an increase in seismic stress can be determined.

Response:

The staff, in a memorandum from E. Case to S. Lawroski dated June 14, 1978, stated that a report on the available seismic margin in the systems for safe shutdown and continued shutdown heat removal at North Anna Power Station Units No. 1 and 2 would be prepared. This report is now scheduled to be sent to the ACRS during December 1978.

APPENDIX B

203rd Meeting, March 10-12, 1977

2. Question:

The approach suggested by the Staff concerning auxiliary system reliability might be adquate, but there is insufficient descriptive information to provide a basis for judgment. It would be useful for the Staff to provide an illustrative example with ficitious data if no meaningful statistics are available as a way of displaying their approach to answering the question.

Response:

An evaluation of the reliability of auxiliary feedwater systems was also requested in a letter from R. F. Fraley to L. V. Gossick, dated July 11, 1978. As stated in the memo from H. R. Denton (September 26, 1978) in response to this request "the Division of Systems Safety is now in the process of initiating a technical assistance contract to evaluate the reliability of various auxiliary systems, such as the component cooling water system, the auxiliary feedwater system for PWR's and the steamline isolation valve leakage control system for BWR's. We expect to complete the preparation of the proposed work scope for this contract early in Fiscal Year 1979. At that time DSS will arrange to brief the Committee on the Program, and our expected schedule for completion".