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Office of Nuclear Reactor Regulation, Director

SUBJECT: Forwards Amend 3 to 820406 TMI-2 lessons learned implementation rept,addressing NUREG-0737 items.

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ARIZONA PUBLIC SERVICE COMPANY

STA. ____________ P.O. BOX 21666 - PHOENIX, ARIZONA 85036

May 24, 1982
ANPP-20853 - WFQ/KWG

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Palo Verde Nuclear Generating Station
Units 1, 2 and 3
Docket Nos. STN-50-528/529/530
File: 82-056-026

Dear Sir:

Arizona Public Service Company (APS), as Project Manager and Operating Agent for Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 & 3, is submitting herewith forty (40) copies of Amendment 3 to the PVNGS Lessons Learned Implementation Report tendered April 6, 1981.

This amendment provides an update to the Lessons Learned Implementation Report.

Sincerely,

E. E. Van Brunt OAR

E. E. Van Brunt, Jr.
APS Vice President
Nuclear Projects
ANPP Project Director

EEVB/KWG/wp
Attachment

cc: P. L. Hourihan
R. Greenfield
E. Licitra
A. C. Gehr

B001
S 5/40 on Shelf

STATE OF ARIZONA)
) ss.
COUNTY OF MARICOPA)

I, A. Carter Rogers, represent that I am Nuclear Engineering Manager of Arizona Public Service Company, that the foregoing document has been signed by me for Edwin E. Van Brunt, Jr., Vice President Nuclear Projects, on behalf of Arizona Public Service Company with full authority so to do, that I have read such document and know its contenets, and that to the best of my knowledge and belief, the statements made therein are true.

A. Carter Rogers

A. Carter Rogers

NOTARY PUBLIC

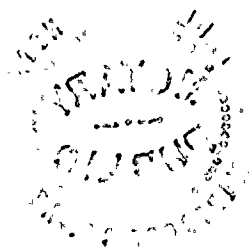
Sworn to before me this 4TH day of MAY, 1982

Paul E. Herndon

Notary Public

My Commission expires:

My Commission Expires Dec. 22, 1985



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I. OPERATIONAL SAFETY

I.A OPERATING PERSONNEL

I.A.1.1 SHIFT TECHNICAL ADVISOR

Position

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The shift technical advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

PVNGS Evaluation

A shift technical advisor (STA) will be provided onsite in addition to the shift supervisor for PVNGS for each shift. The STA 3 will serve all three PVNGS units. The duties of the STA will include:

- Diagnose accidents and off-normal events for their significance to reactor safety and advise the shift supervisor.

- Incorporation into the onsite Independent Safety Engineering Group (see section I.B.1.2).

3 Organizationally, the STA will report through the supervising engineer of the Independent Safety Engineering Group to the Operations Engineering Supervisor who is independent of operations.

STA training, qualifications, and selection criteria are discussed in FSAR Section 13.2.1.3.2. STA requalification training will be conducted as described in amended FSAR Section 13.2.2.2.3. Facility Technical Specification 6.2.4 and Table 6.2-1 will be proposed, further describing the station and duties of the STA.

I.A.1.3 SHIFT MANNING

Position

(1) Limit Overtime

Administrative procedures shall be established to limit maximum work hours of all personnel performing a safety-related function. 2

(2) Minimum Shift Crew

The minimum shift crew for a unit shall include three operators, plus an additional three operators when the unit is operating. Shift staffing may be adjusted at multi-unit stations to allow credit for operators holding licenses on more than one unit.

In each control room, including common control rooms for multiple units, there shall be at all times a licensed reactor operator for each reactor loaded with fuel and a senior reactor operator licensed for each reactor that is operating. There shall also be onsite at all times, an additional relief operator licensed for each reactor, a licensed senior reactor operator who is designated as shift supervisor, and any other licensed senior reactor operators required so that their total number is at least one more than the number of control rooms from which a reactor is being operated.

PVNGS Evaluation

1. Limit Overtime

PVNGS administrative procedures shall, by fuel load, pro-
2 vide provisions limiting maximum hours worked by personnel performing a safety related function to the guidelines of NRC Generic Letter No. 82-02:

a. An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period (all excluding shift turnover time).

3 c. A break of at least eight hours should be allowed between work periods (including shift turnover time).

d. The use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Recognizing that very unusual circumstances may arise requiring deviation from the above guidelines, such deviation shall be authorized by the Manager of Nuclear Operations or his designee, or higher levels of management. The paramount consideration in such authorization shall be that significant reductions in the effectiveness of operating personnel would be highly unlikely.

In addition, procedures encourage licensed operators at the controls to be periodically relieved and assigned to other duties away from the control board during their tour of duty.

The personnel effected by this requirement will be senior reactor operators, reactor operators, radiation protection technicians, auxiliary operators, I & C technicians and key maintenance personnel. 3

2. Minimum Shift Crew

The minimum shift crew for a unit is discussed in FSAR Section 13.1.2.3 and FSAR Table 13.1-2 and meets the above requirements. 3

PVNGS administrative procedures will by fuel load provide provisions governing required shift staffing. 2

I.B OVERALL ORGANIZATION

I.B.1.2 INDEPENDENT SAFETY ENGINEERING GROUP

Position

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities, including maintenance, modifications, operational problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

PVNGS Evaluation

In response to the recommendations of NUREG-0737, PVNGS has made changes to the plant staff organization which provide additional assurance that PVNGS is operated in a safe manner. These organizational changes involve the Shift Technical Advisors (STA), as well as the Independent Safety Engineering Group (ISEG) and feedback of operating experience function. The following is a description of the the organizations and staffing changes.

Independent Safety Engineering Group Organization

As described in NUREG-0737, recently proposed revisions to
3 Regulatory Guide 1.33, and INPO's recommendations on this subject, the independent safety review, operating experience evaluation, and Shift Technical Advisor accident assessment functions are directly related and overlap in many areas. Accordingly, we have combined the STA responsibilities for all three units and the ISEG functions into a single onsite organization. This group has a close relationship with the Safety Audit Committee, our staff organization, which performs company independent safety reviews.

In addition to consolidating the closely related functions, this arrangement has the following advantages:

a. An awareness of industry operating experience will be an important aid in STA accident assessment.

b. The ISEG duty "surveillance of plant activities to provide independent verification that these activities are performed correctly and human errors are reduced as much as possible" gains the STA familiarity with the equipment and a feeling for what types of equipment failures and human errors are most likely to occur.

c. Preparation of responses to IE circulars and information notices and other reports on plant problems gives the STA's exposure to management philosophy, an appreciation for the rigorous approach required, and also gives management a chance to evaluate STA performance prior to an accident environment. 3

d. STA's on duty need more work to do than accident assessment. ISEG duties enhance the STA job and provide a well organized set of work assignments.

e. STA training and responsibility form a good basis for ISEG decisions.

The Independent Safety Engineering Group will not replace either the Safety Audit Committee or the Plant Onsite Review Committee. Its members support and may receive technical direction from the Safety Audit Committee. Administratively, they report to the Engineering and Technical Services Manager who is independent from the responsibilities for day-to-day operations. While on duty as STA, they advise the shift supervisor and report functionally through the station organization.

APS has issued a policy document requiring that: (1) reports of the ISEG dealing with safety issues be sent to the Chairman of the Safety Audit Committee (SAC), and (2) safety issues raised by the ISEG are to be reviewed and resolved by the Chairman of the SAC. The policy provides an appeal path for resolution of potential differences between the Chairman of SAC and the Vice President of Electric Operations.

Work Schedule and Function

Our plan is to have the STA/ISEG personnel stand watch on a 24-hour duty day basis. Thus, they will be asleep at times while on duty, but will be available in the Control Room on short notice. As we gain more experience with this arrangement,
3 we will continually reassess the work schedule and make modifications needed to provide the most effective arrangement.

As STA, the primary responsibility is to provide technical assistance to the shift supervisor during an off-normal event. When on duty, but not assisting the shift supervisor, and when off duty, these personnel will perform the functions of the ISEG as listed below:

a. See item I.C.5 for a description of the operating experience information evaluation program.

b. Independent Evaluation and Surveillance of Plant Activities

A wide range of plant activities, including operations, maintenance, and modifications are monitored during the actual performance of the work to evaluate the technical adequacy of the methods used, recommend equipment changes, and aid in the establishment of programmatic requirements for plant activities. This surveillance provides independent verification that these activities are performed correctly and reduces the potential for human error as far as possible.

c. Other duties involving the safe operation of the plant as directed by the Safety Audit Committee. 3

The combined STA/ISEG Organization will be staffed with persons who meet the qualifications of FSAR Section 13.2.1.3.2 as STA, and the organization will have at least five members with at least three years average level of nuclear power plant experience.

Facility Technical Specification 6.2.3 will be proposed to provide ISEG function, composition, responsibilities, and authority.

The C-E Owners Group held a meeting with the NRC Division of
Systems Integration and Human Factors Safety on January 30, 1981,
to discuss the process being used for revision of the emergency 1
procedure guidelines. The revised emergency procedure guide-
lines were submitted to the staff on June 30, 1981 as CEN-152
in addition to the submittal of CEN-156 titled Emergency Pro-
cedure Guideline Development.

The NRC provided comments on these documents in a July 24, 1981 2
meeting and in a letter to C-E Owners Group dated September 15,
1981. The C-E Owners Group is in the process of reviewing these
guidelines in light of the NRC comments.

PVNGS intends to submit a Procedures Generation Package, in
accordance with Section 6.0 of Draft NUREG-0899, to the NRC for
staff review following NRC approval of C-E Owner's Group emer-
gency procedure guidelines. A target submission date of 3
August 1, 1982 has been established for this package. Emergency
operating procedures will be developed and implemented in accord-
ance with this package and will be ready for NRC onsite review
60 days prior to fuel load.

I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

Position

In accordance with Task Action Plan I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

(1) Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

(2) Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures; operating orders);

(3) Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians)

or otherwise provide means through which such information can be readily related to the job functions of the recipients;

(4) Provide means to assure that affected personnel become aware of and understand information of sufficient importance that should not wait for emphasis through routine training and retraining programs;

(5) Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

(6) Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

(7) Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

PVNGS Evaluation

An operating experience review program is being developed for PVNGS which will establish the responsibilities and methodologies for reviewing the operating experience of PVNGS and other nuclear plants. PVNGS will participate in the Institute of Nuclear
3 Power Operations (INPO) Significant Event Evaluation and Information Network (SEE-IN) as discussed in NRC Generic Letter

No. 82-04. The program and implementing procedures will be developed in accordance with this requirement and will be in effect prior to Unit 1 fuel load. 3

E. Radiation Monitoring

1. Response of Process and Area Monitors to severe damages; behavior of detectors when saturated; method for detecting radiation readings by direct measurement at detector output (overranged amplifier); expected accuracy of detectors at different locations; use of detectors to determine extent of core damage.

2. Methods of determining dose rate inside containment from measurements taken outside containment.

F. Gas Generation

1. Methods of H_2 generation during an accident; other sources of gas (Xe, Kr); techniques for venting or disposal of non-condensables.

2. H_2 flammability and explosive limit; sources of O_2 in containment or Reactor Coolant System.

(2) Complete Training

The course shall be developed and these personnel shall participate prior to fuel loading of unit one.

PVNGS Evaluation

(1) Develop Training Program

A course will be developed to train shift technical advisors and operating personnel through the operations

chain to the licensed operators in the use of installed plant systems to control or mitigate an accident in which the core is severely damaged. The training program will include the topics suggested in the H.R. Denton letter of March 28, 1980. This training will consist of approximately 40 hours of formal classroom presentations by either a private consultant or the engineering staff. Operators
3 will acquire the theoretical basis for these actions in the academic programs and the practical application during the simulator training course. The total training time will exceed 80 hours.

Managers and technicians in the Instrumentation and Control (I & C), radiation protection and chemistry sections will receive training commensurate with their responsibilities that meets the requirements of the H.R. Denton letter of
3 March 28, 1980.

(2) Complete Training

3 The above training will be completed prior to fuel load operation.

II.F INSTRUMENTATION AND CONTROLS

II.F.1 ADDITIONAL ACCIDENT-MONITORING INSTRUMENTATION

A human factor analysis will be performed to ensure that the 1
displays and controls added for additional-accident accomplish
this monitoring do not increase the potential for operator
error (see section I.D.1). Installation will be completed 2
prior to fuel load.

II.F.1.1 ATTACHMENT 1, NOBLE GAS EFFLUENT MONITOR

Position

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

(1) Noble gas effluent monitors with an upper range capacity of 10^5 µCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA)) concentrations to a maximum of 10^5 µCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

PVNGS Evaluation

FSAR section 11.5 provides detailed descriptions of the effluent
1 monitors installed at Palo Verde Units 1, 2 and 3. This includes
the additional monitors that have been added specifically to
3 address NUREG-0737 and Reg Guide 1.97, Rev 2 requirements
2 for radiation monitoring. Installation and calibration will be
3 completed by fuel load. A description of the calibration
sources, frequency of calibration, and technique is pro-
vided in FSAR Table 11.5.1 and Sections 11.5.2.1.6.2 and
2 11.5.2.1.6 respectively. The instrumentation is described in
detail in FSAR Table 11.5-1. The outputs of the effluent
monitor and flow meter will be in μCi/cc and cc/h respectively
and when obtained need only be multiplied to obtain μCi/h.
3 Sampling of effluents will meet the criteria of ANSI N13.1 1969
as discussed in FSAR Sections 11.5.2.1.1.7.2.2 and 11.5.2.2.1.
Monitors are designed to meet a 90% efficient level for partic-
2 ulates and 90% efficiency for iodine as required by NUREG-0737
Table II.F.1-2. They are also designed to conform with design
basis shielding envelopes for sampling media as discussed in
FSAR Section 12.1.2.4 and item II.B.2. Monitors are designed
to allow personnel to remove, replace, and transport sampling
media without exceeding the criteria of GDC19 of 5 rem whole-
3 body and 75 rem to the extremities.

Each process or effluent channel includes a sampling assembly which consists of a sampler and the associated piping, fittings, and other components as required to transport the sample

through the system. The sampling assembly is a closed, sealed
system and includes a sampling pump, valves, interconnecting
piping, filters, fittings, flow and pressure transducers, and
other local control and instrumentation elements as required.
Samplers, with the exception of the condenser vacuum pump/gland
seal exhaust particulate-iodine sampler, house radiation detec-
tion equipment and check source(s). Sampler piping and
connections are welded except where maintenance considerations
make flanged or Swagelok joints necessary. Sampler outlet
piping connections are located to minimize cleaning requirements
and background buildup due to the adherence of radioactive
particles to the sampler walls. For liquid samplers, welding
of pressure-containing components is performed in accordance 3
with ANSI B31.1. For ESF monitors, welding of pressure-
containing components is performed in accordance with
AWS D1.1-1972 (with 1973 revisions). Welding of other equipment
is performed in accordance with industry standards.

For liquid and process channels, the sampler is a lead-shielded steel chamber. For particulate and iodine channels, the sampler is a lead-shielded filter assembly. Four π shielding is furnished for all process and effluent detectors.

Airborne particulate and iodine monitors and samplers, with the exception of the containment building atmosphere monitor, sample isokinetically in accordance with the principles and methods of ANSI N13.1-1969, Guide to Sampling Airborne

Radioactive Materials in Nuclear Facilities. The particulate and iodine sample flow is maintained constant over the normal expected range of filter paper and/or charcoal cartridge differential pressure by an automatic control system. Local flow indication and high- and low-flow alarm signals are provided. These signals actuate local alarms and the channel failure alarms. Particulate samplers (except the main con-
3 denser air ejector low range) are moving paper filter type and incorporate microcomputer-controlled step advance and feed failure channel failure alarm. Sampling assembly fittings are provided which allow grab sampling of the monitored airstreams.

A flow-integrating elapsed sample volume indicator is provided downstream of each particulate and/or iodine channel. It has a local digital readout and is resettable to zero. A human
2 factor analysis was performed as discussed in item I.D.1.

A. Wide-Range Effluent Monitor

In order to cover the dynamic range required, a normal range monitor is used with a high range monitor. One decade is used for overlap when switching between monitors. Both monitors sample using controlled isokinetic flows. Their combined range is $10^{-6}\mu Ci/cc$ to $10^{+5}\mu Ci/cc$. These monitors are located on the plant vent, main condenser/gland seal exhaust, and the fuel building vent. In addition to noble gas monitoring capability, these instruments have separate particulate and iodine sample chambers for both the low and high range, and use charcoal
2 or Silver-Zeolite cartridges in conjunction with a portable

pump. All cartridges will be removed to the counting laboratory
for gamma spectrum analysis. High range monitor iodine samples
are provided with a lead shield. Procedures will be developed to
define ALARA concepts for removal, transport and analysis. These
monitors have complete digital readout and control from the Health
Physics Office and the main control room. The high range monitors
automatically switch to a new particulate/iodine cartridge pair
when the current cartridge reaches a preset radiation level.
Filter materials used minimize absorption of noble gases.
Samples are preconditioned as necessary to assure accurate
results without damaging the sample assemblies. Each monitor
is controlled by a remote microprocessor. This microprocessor
is linked by a "daisy chain" to a minicomputer which provides 1
multiple informational displays on request by the operator.
A dedicated alarm status line is maintained on the CRT display.
This status line does not move with each change of CRT displays.
Thus alarms are provided regardless of the status of the
displays in the Health Physics Office and Main Control Room.
Monitors are provided with an open structural construction
that provides for easy maintenance and good heat dissipation.
Backup battery power is provided to assure continued micro-
processor memory during a loss of external power sources.
Multiple detectors are used to achieve the dynamic range
required. Hard copy readouts are available from dedicated
printers in the Health Physics office and the control room.

B. Main Steam Line Monitor

1 One area monitor with a collimating lead shield is mounted
adjacent to each main steam line in the Main Steam Support
Structure approximately one foot upstream of the atmospheric
3 dump valves. Refer To FSAR Figure 12.3-2. These monitors
measure direct dose rates from the main steam line to identify
effluent from the atmospheric dump, main steam relief valves,
and auxiliary feedwater pump discharge. An extra 2 inches
of shielding is placed on the containment side of the detector
shield. There are a total of 4 detectors with one remote
microprocessor for each 2 detectors. The ion chamber covers
a range from 1mr/h to 10^7mr/h. The microprocessor is linked
by a "daisy chain" to a minicomputer which provides multiple
informational displays at the request of the operator.
Alarms are provided regardless of the status of the displays
1 in the Health Physics Office and the Control Room. Hardcopy
readouts are provided from dedicated printers in the Health
Physics Office and the Control Room. Backup battery power is
provided to assure continued microprocessor memory during a loss
of external power sources. The detector is designed to operate
in a post-accident environmental condition with a background
of 10 R/h.

C. Meeting the NRC Requirement

These monitors operate in conjunction with the other monitors as
discussed in FSAR section 11.5 and fulfill the requirements as

outlined in NUREG-0737 and Regulatory Guide 1.97, Rev 2. 3
Installation will be completed prior to fuel load. 2

II.F.1.2 ATTACHMENT 2, SAMPLING AND ANALYSIS OF PLANT EFFLUENTS

Position

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by adsorption on charcoal or other media, followed by onsite laboratory analysis.

PVNGS Evaluation

PVNGS response to this item is included in the evaluation of section II.F.1.1 requirements.

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Data acquisition of the two independent meteorological system signals may be accomplished for projected dose calculations, CRT displays and remote data transmission. 1
The same capability is provided for the nuclear data link.
As independent, redundant and validated signals are used, 2
the system can achieve a non-availability of 0.01.

Below is a summary table of the sensors used in each independent meteorological system to monitor the environmental 1.
parameters, and the common data processing available.

Sensor / Location	200' Wind Speed and Wind Direction Sensor	35' Wind Speed and Wind Direction Sensor	200' & 35' Aspirated Temperature Sensor	35' Aspirated Dewpoint Sensor	Rainfall Monitor Located at Ground Level	2
Tower System A	Provided	Provided	Provided	Provided	Provided	1.
Tower System B	Provided	Provided	Provided	Not Provided	Not Provided	
Digital & Analog Process-ing	Provided	Provided	35' Ambient Only 200'-35' ΔT	Provided	Provided	2

The environmental parameters monitored by each independent tower system permits highly accurate and reliable meteor- 1
ological data necessary to cover all data for the Pasquill stability classes and transport protections needed for the PVNGS site.

The auxiliary analog information for each tower system is provided on analog recorders located at the meteorological station in the respective tower system equipment trailers. Analog information for each tower system is converted to digital data and transmitted by two serial links to the plant site where the data is reduced to fifteen-minute
1 and hourly average meteorological parameters, and where all effluent and environmental parameters are recorded and available for time history displays in the control room, emergency response facilities, and at external locations. Serial data is also provided to the radiation exposure management system to meet the reporting requirements of
2 Regulatory Guide 1.21. In addition, the signal is available to the multi-channel processing system for offsite and projected dose calculation, technical support center and the emergency operations facility CRT displays.

The models used for providing the estimates of offsite exposure are:

1 - Plume exposure - Class A model will be used with 15 minute average meteorology data for initial transport and diffusion estimates within 15 minutes following the classification of the incident.

3 - Ingestion Zone - A modified Class A model will be used for relative concentration of the plume emer-
1 gency planning zone (EPZ) and ingestion EPZ.

III.D RADIATION PROTECTION

III.D.1.1 INTEGRITY OF SYSTEMS OUTSIDE CONTAINMENT LIKELY TO CONTAIN RADIOACTIVE MATERIAL FOR PRESSURIZED-WATER REACTORS AND BOILING-WATER REACTORS

Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

PVNGS Evaluation

3 1. Design Review

Refer to CESSAR Appendix B, Item III.D.1.1. In addition, a PVNGS design review was performed on the system below to assure that potential radioactive release paths following a serious transient or accident is reduced to as-low-as-reasonably achievable (ALARA) levels.

1 A. Shutdown Cooling System (SCS)

The existing design incorporates all-welded piping. Vent and drain lines throughout the system are capped when not in use. Relief valves on the system relieve to the equipment drain tank (a tank designed to accept radioactive fluids). The leakage from the LPSI pump seals and system valve stems is ALARA. Potential leakage from the SCS into the essential cooling water system (through the shutdown heat exchanger) can be detected during normal operation by installed radiation monitoring.

1 B. Containment Spray Recirculation System (CS)

The existing design incorporates all-welded piping. Vent and drain lines throughout the system are capped when not in use. Relief valves on the system (external to the containment) relieve to the equipment drain tank. The leakage from the CS pump seals and system valve stems is ALARA. Potential leakage during normal operation from the CS into the essential cooling water system (through the shutdown heat exchanger) can be detected by installed radiation monitoring.

C. <u>CVCS Charging and Letdown System</u>

The existing design incorporates all-welded piping. The letdown system is isolated upon CIAS and SIAS. Relief valves on the system relieve to the equipment drain tank.

The leakage from the CVCS charging pumps (positive displacement pumps) and other system equipment is ALARA as they are hard-piped to drains. The nuclear cooling water system is monitored for potential leakage from the CVCS through the letdown heat exchanger during normal operation.

D. <u>Sampling System</u> 1

The existing design of the normal sampling system incorporates "Swagelok" connections, however, the design will be upgraded to all-welded piping for sections which would come into contact with highly radioactive fluids. The system is isolated upon CIAS and SIAS. Relief valves relieve to the equipment drain tank. Leakage from the system is also minimized by the small size of the lines.

3 The post-accident sampling system will also be constructed of all-welded piping, except within cabinets.

E. High-Pressure Injection Recirculation (HPSI)

The design incorporates all-welded piping. Relief valves on the system (external to the containment) relieve to the equipment drain tank. The vent and drain lines throughout the system are capped when not in use. Leakage from the HPSI pump seals and system valve stems is as-low-as reasonably achievable. Miniflow connections to the refueling water tank (RWT) are isolated upon Recirculation Actuation Signal (RAS). Manual cross over valves to the CVCS are normally locked shut.

1 F. Waste Gas System

The waste gas system is isolated from the containment upon CIAS. (The normal vent path from the reactor drain tank (RDT) and the reactor head vent system is isolated.) By design, the introduction of highly radioactive fluids to the system is precluded.

2 As part of the system testing program, each of the above systems is hydrostatically tested to 150% normal operating pressure per the requirements of ANSI B31.1, Summer 1976 Addendum for ANSI B31.1 piping systems, and to 125% normal

operating pressure per the requirements of ASME Boiler & Pressure Vessel Code, Section III, 1977 Edition, for ASME piping systems.

2. Leakage Reduction Program 3

PVNGS will institute a program to maintain leakage rates of systems outside containment to as low as practical which consists of the following: 2

A. Systems Included in the Program 3

1. High pressure safety injection system (recirculation portion only).
2. Low pressure safety injection system (shutdown cooling portion only).
3. Reactor coolant sampling system (post-accident sampling piping only). 2
4. Containment spray system.
5. Radioactive waste gas system (post-accident sampling return piping only).
6. Liquid radwaste system (post-accident sampling return piping only).
7. Containment combustible gas and atmospheric sampling system (hydrogen, monitoring subsystem and post-accident sample piping associated with this function). 3

3 B. Systems excluded from the program: (They will not preclude any option of cooling the reactor core nor will they prevent the use of needed safety systems).

1. Radioactive liquid waste system, except as
2 discussed above.

2. Radioactive waste gas system, except as discussed above. (The system is not required for use post-accident.)

3. Reactor coolant letdown system, except for
3 portions required for post-accident sampling described in FSAR section 9.3.2.2.2. (The system is not required to function post-accident. The plant can be brought to a cold shutdown condition without the letdown system. The letdown system is isolated on SIAS and CIAS.)

4. Reactor coolant pump seal bleed-off system. (The
2 system is not required to function outside containment post-accident. The seal bleed-off system is isolated outside containment on CIAS. The system remains isolated post-accident. If seal bleed-off is required post-accident, pressure in the seal bleed-off header will increase and the header relief valve will lift providing a flow path to the reactor drain tank.)

5. Charging System. (The charging system under post-accident conditions does not contain radioactive fluid since the letdown system is isolated as dis- 2
cussed in item 3 above. The charging system takes suction from the refueling water tank.)

6. Fuel Pool Cooling system (FPC). (The FPC is 3
normally isolated from potentially highly con-
taminated systems by double, locked shut 2
isolation valves.)

C. Program Features 3

Immediate leak reduction measures: The program will
consist of periodic monitoring of the systems during 2
operation and inservice leak testing. Leaks will be 3
identified and corrective maintenance performed. 2

1. Vent and drain lines will be capped to prevent 3
release due to seal leakage. 2

2. The packing of valves (except Kerotest which is 3
a packless, stainless steel diaphragm valve) in the scoped liquid systems will be inspected for leakage or evidence of leakage such as boric acid 2
accumulation. Maintenance will be performed on the packing of liquid system valves identified as requiring work.

3. The seals and packing on pumps in the scoped 3
liquid systems will be inspected for leakage or 2
signs of leakage.

4. Valves, fittings, and compressor seals in the scoped gaseous systems will be checked for leakage.
2 Maintenance will be performed on gas system valves and instrument fittings identified during leak tests as requiring work.

5. Systems and subsystems identified in 2A will be leak tested prior to exceeding 5% power and on an interval not to exceed the period between refueling outages. Test records including measured leak rates will be maintained at PVNGS for NRC review. A report including the measured leak rates will be submitted for NRC staff review prior to operation above 5% power. Leak rate test techniques will include:

   a. Liquid Systems

3 A visual examination will be performed on items 1 through 3 of paragraph 2C above with the system at or near operating pressure. If leakage is identified during these examinations, an integrated leakage rate will be determined by monitoring the applicable sump and tank levels. For sumps and tanks that do not contain a level indicator, the levels will be determined by physical measurement. In addition, the local leak rate tests performed on isolation valves will be utilized for the portion of each system located between the containment and

the isolation valves, if practical. These tests will be performed in accordance with written Station Manual procedures.

b. Gas Systems

The leakage will be determined by detecting gas leakage at individual valves, fittings, seals, and bolted connections with the system at or near operation pressure. Leakage will be detected by use of acoustic, bubble, or equivalent method (such as a tracer gas method). In addition, the local leak rate tests performed on isolation valves will be utilized for that portion of each system located between the containment and the isolation valves, if practical. These tests will be performed in accordance with written Station Manual procedures. 3

The PVNGS design was reviewed to confirm that the design and construction of PVNGS systems minimize unplanned releases of radioactivity including the related incidents identified in NRC letter dated October 17, 1979 to all operating nuclear power plants. The following summarizes that review:

Radioactive liquid atmospheric tanks are provided with overflows with either no isolation valve or a locked-open valve. Overflow lines have loop seals and are routed to appropriate radioactive building sumps.

The sump liquid is routed to the LRS holdup tanks. Overflow lines from the refueling water tank and the LRS concentrate monitor tanks are heat-traced to prevent plugging. Radioactive liquid pressurized tanks with the exception of the volume control tank and reactor drain tank are provided with relief lines routed to the appropriate sumps. A summary of the overflow provisions for the radioactive tanks is provided in table III.D.1.1-1.

Storm drains are located away from areas with a high potential for radioactive spills. No storm drains exist in the immediate vicinity of the Containment, Auxiliary, or Radwaste Buildings.

Radioactive pumps are generally located in isolated compartments whose drains are designed to catch all potential leakage. These
3 drains are routed to the appropriate radioactive building sump. In addition, certain pumps whose potential for radioactive leakage is greatest are equipped with drip pans with lines hard-piped to the associated building sump. A summary of the radioactive pumps and their leakage provisions is given in table III.D.1.1-2.

Radioactive valves are located in shielded compartments such as valve galleries equipped with floor drains that are designed to collect all potential valve leakage. These drains are routed to appropriate building sumps.

Radioactive tanks located inside the Auxiliary and Radwaste Buildings are located in compartments with curbs to contain tank leakage. These compartments are also equipped with floor drains routed to the appropriate radioactive building sump. Outside liquid radwaste tanks are surrounded by a dike sufficient to hold the contents of a tank rupture. Outside CVCS tanks are concrete tanks with steel liners. The concrete tanks will retain potential liner leakage.

The hot lab, cold lab, decontamination area, and sample station 3
are equipped with floor drains routed to the non-ESF sump. There are no piping systems between units which could become contaiminated.

Based on this discussion the North Anna-type event is not expected to occur at PVNGS.

Table III.D.1.1-1

RADIOACTIVE TANKS OVERFLOW AND LEAKAGE PROTECTION

P&ID	Tank	Atmospheric or Pressure Vessel	Overflow or Relief	Overflow or Relief Line	Tank Location	Curb or Enclosed Compartment	Comments
CHP-001	Volume Control Tank	PV	Relieves to vent gas surge header	N-214-HCDA-3/4"	Auxiliary Bldg 120' level	Enclosed Compartment	
CHP-002	Refueling Water Tank	ATM	Overflow to holdup tank sump	N-134-HCDA-6"	Outside of fuel bldg	Concrete w/ steel liner	Overflow line is heat-traced.
CHP-003	Reactor Makeup Water Tank	ATM	Overflows to holdup tank sump	N-381-HCDA-3"	Outside of fuel bldg	Concrete w/ steel line	
CHP-001	Radwaste Crud Tank	PV	Relieves to non-ESF sump	N-533-GCDA-2"	Auxiliary Bldg 100' level	4" curb	
CHP-003	Reactor Drain Tank	PV	Vents to gas surge tank	N-281-HCDB-2"	Containment 80' level	-	
CHP-003	Equipment Drain Tank	PV	Relieves to non-ESF sump	N-347-HCDB-1"	Auxiliary Bldg 40' level	-	
CHP-003	Holdup Tank	ATM	Overflows to holdup tank sump	N-353-HCDA-3"	Outside of fuel bldg	Concrete w/ steel liner	
LRP-001	Low TDS Holdup Tank	ATM	Overflows to radwaste bldg sump	N-014-HCDA-6"	Outside of Radwaste Bldg	Enclosed compartment	
LRP-001	High TDS Holdup Tanks	ATM	Overflows to radwaste bldg sump	N-229-HCDA-4"	Outside of Radwaste Bldg	Enclosed compartment	
LRP-001	Chemical Drain Tanks	ATM	Overflows to aux bldg sump via a funnel drain	N-067-HCDA-3" N-206-HCDA-2"	Auxiliary Bldg 51'-6" level	6" curb	
LRP-002	Concentrate Monitor Tanks	ATM	Overflows to radwaste bldg sump	N-195-HCDC-2" N-219-HCDC-1"	Radwaste Bldg 100' level	6" curb	Overflow lines are heat-traced.
LRP-002	Recycle Monitor Tanks	ATM	Overflows to radwaste bldg sump	N-183-HCDA-3" N-205-HCDA-3"	Outside of Radwaste Bldg	Enclosed compartment	
SRP-001	High Activity Spent Resin Tank	PV	Relieves to radwaste bldg sump	N-027-HCDA-2"	Radwaste Bldg 100' level	Curb	
SRP-001	Low Activity Spent Resin Tank	PV	Relieves to radwaste bldg sump	N-016-HCDA-2"	Radwaste Bldg 100' level	Curb	
SRP-002	Waste Feed Tank	ATM	Overflows to radwaste bldg sump via funnel drain	N-204-HCDC-3/4"	Radwaste Bldg 100' level	Enclosed Compartment	

3

3

Table III.D.1.1-2

RADIOACTIVE PUMPS LEAKAGE PROVISIONS

3

P&ID	Pump	Drain Pan Drain Line	Location	Comments
CHP-001	Crud Pump	N-554-HCDA-1". Drains to non-ESF sump	Auxiliary Bldg 100' level	None
CHP-002	Charging Pumps	N-245-HCDB-1" N-246-HCDB-1" N-247-HCDB-1". Drain to recycle drain header.	Auxiliary Bldg 100' level	None
CHP-002	Boric Acid Makeup Pumps	N-449-XCDA-1/2" N-453-XCDA-1/2". Drain to non-ESF sump.	Auxiliary Bldg 70'-0' level	Equipped with a gland seal loop off the process flow
CHP-003	Reactor Makeup Water Pumps	No drip pan. Drain line off gland seal to holdup tank sump	Auxiliary Bldg 70' level	Equipped with a gland seal loop off the process flow
CHP-003	Reactor Drain Pumps	N-476-XCDA-1" N-479-XCDA-1". Drain to a funnel drain routed to non-ESF sump	Auxiliary Bldg 40' level	None
CHP-003	Holdup Pumps	N-482-XCDA-1/2" N-488-XCDA-1/2". Drain to holdup tank sump	Auxiliary Bldg 40' level	Equipped with a gland seal loop off the process flow
LRP-001	LRS Holdup Pumps	N-031-HCDA-1" N-032-HCDA-1" N-033-HCDA-1". Drain to radwaste bldg sump	Radwaste Bldg 100' level	None 3
LRP-001	Chemical Drain	N-079-HCDA-1" N-082-HCDA-1" Drain to a funnel drain routed to radwaste bldg sump	Auxiliary Bldg 40'-0" level	None
LRP-002	Concentrate Monitor Tank Pumps	N-117-HCDC-1" N-620-HCDC-1" Drain to radwaste bldg sump	Radwaste Bldg. 100' level	Drain line is heat traced
LRP-002	Recycle Monitor Pumps	N-186-HCDA-1". Drains to radwaste bldg sump	Radwaste Bldg 100' level	None
PCP-001	Fuel Pool Cleanup Pumps	No drip pan	Fuel Bldg 100' level	Equipped with a gland seal loop off the process flow which drains to fuel building sump
PCP-001	Fuel Pool Cooling Pumps	No drip pan	Fuel Bldg 100' level	Equipped with a gland seal loop off the process flow which drains to fuel building sump
SRP-001	Resin Transfer/ Dewatering Pump	N-081-HCDA-1". Drains to a local stub-up routed to radwaste bldg sump	Radwaste Bldg 100' level	None
SRP-002	Waste Feed Pump	N-068-HCDC-1". Drains to a local stub-up routed to radwaste bldg sump	Radwaste Bldg 100' level	None

III.D.3.3 IMPROVED INPLANT IODINE INSTRUMENTATION UNDER ACCIDENT CONDITIONS

Position

(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

PVNGS Evaluation

Prior to fuel load, procedures will be developed for determin-
ing airborne iodine concentration. Silver-Zeolite or charcoal 2
cartridges will be used in conjunction with a portable pump.
The cartridges will be removed to the counting laboratory for
gamma spectrum analysis. Cartridges will be purged with clean 3
bottled compressed air prior to counting. A low background
counting facility is available on the site. The results 2
of airborne concentration can be obtained within 15 to
30 minutes after collection of iodine on filtered cart- 3
ridges. Procedures will also define ALARA concepts for 2

removal, transport, and analysis of filter cartridges. There
2 will be at least three such portable airborne monitors available at each unit which meet the NUREG-0737 recommendations.

1 PVNGS response to this item is included in the evaluation of section II.F.1 requirements.

IV. RESPONSES TO NRC REQUESTS FOR INFORMATION

QUESTION IV.1 (NRC I&E Question 20) (I.B.1.2)

Please indicate your commitment that the Independent Safety Engineering Group (ISEG) which you plan to establish per your letter of April 6, 1981 will meet all of the guidance provided in Section I.B.1.2 of NUREG-0737 (November 1980). Also, please commit that the members of this group will satisfy the qualification requirements for Staff Specialists as defined in Paragraph 4.7.2, ANS 3.1-1978; and that the group will provide to management, no less frequently than monthly, a summary of their activities to advise management on the overall quality and safety of operation. 3

RESPONSE: Refer to amended section I.A.1.3. A general description of our organization which combines the STA program with the ISEG program is given in amended section I.A.1.1. and I.B.1.2.

QUESTION IV.2 (NRC I&E Question 21) (I.A.1.3)

Please indicate your commitment that the PVNGS procedures limiting overtime (as committed to in your letter of April 6, 1981) will conform to the guidance contained in Section I.A.1.3 of NUREG-0737 (November 1980).

RESPONSE: The response is provided in amended section I.A.1.3.

QUESTION IV.3 (NRC I&E Question 24) (I.A.1.1)

Please describe where the Shift Technical Advisor Group and the Independent Safety Engineering Group are located in the APS Nuclear Operations organization.

RESPONSE: Refer to amended sections I.A.1.1 and I.B.1.2.

QUESTION IV.4 (NRC I&E Question 25) (I.B.1.2)

Please furnish copies of the following procedures you committed to prepare in response to the TMI Task Action Plan Items listed below:

I.A.1.2	-	Shift Supervisor Administrative Duties
I.A.1.3.1	-	Overtime
I.A.1.3.2	-	Shift Manning and Movement of Key on-Shift Personnel
I.B.12	-	Independent Safety Engineering Group (Group Charter)
I.C.2	-	Shift Relief and Turnover Procedures
I.C.3	-	Shift Supervisor Responsibilities
I.C.4	-	Control Room Access
I.C.5	-	Procedures for Feedback of Operating Experience to Plant Staff

3

RESPONSE: These procedures will be available onsite for review six months prior to fuel load.

QUESTION IV.5 (F.J. Miraglia letter dated January 8, 1982) (III.D.1.1)

Your response to our earlier request for additional information relating to this action plan item has not addressed the following items. These should be addressed:

a. Leak test methods for liquid and gaseous systems.

b. Applicability to Palo Verde of North Anna and related incidents (identified in NRC's letter dated 10/17/79 to all operating nuclear power plants).

c. Measured actual leak rates from all applicable systems with the system in operation (at this time, at least a commitment must be made that these will be submitted according to the schedule given in NUREG-0737). 3

d. Frequency of the periodic integrated leak tests.

e. Major features of the continuing Leak Reduction Program.

f. Leak testing for the containment sampling system.

g. Leak testing for Residual Heat Removal System.

RESPONSE:

a. The response is given in amended section III.D.1.1.

b. The response is given in amended section III.D.1.1.

c. The response is given in amended section III.D.1.1.

d. The response is given in amended section III.D.1.1.

e. The response is given in amended section III.D.1.1.

f. The response is given in amended section III.D.1.1.

g. The function of a "residual heat removal system" is performed by the shutdown cooling system which consists
3 of portions of the high pressure and low pressure safety injection and containment spray systems, which are included in the leakage reduction program described in section III.D.1.1(2).