U. S. NUCLEAR REGULATORY COMMISSION

REGION V

Report Nos. 50-528/91-53, 50-529/91-53 and 50-530/91-53

License Nos. NPF-41, NPF-51, and NPF-74

Licensee: Arizona Nuclear Power Project (ANPP)
P. O. Box 52034
Phoenix, AZ 85072-3999

Facility name: Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, & 3

Inspection at: PVNGS Site at Wintersburg, AZ

Inspection conducted: December 30 - 31, 1991, through January 2 - 3, 1992

Inspection by: G.P. Yuhas for 2/4/92
L. C. Carson II, Reactor Radiation Specialist Date Signed

Approved by: GP Yuhas 2/4/92
G. P. Yuhas, Chief Date Signed
Reactor Radiological Protection Branch

Summary:

Areas Inspected:

Routine unannounced inspection of the licensee's process and effluent radiation monitoring systems (RMS) and followup on previous inspection findings. Inspection procedures 84524, 84724, and 92700 were used.

Results:

The licensee continues to evaluate and improve their radiation monitoring systems. Their use of the incident investigation process in connection with these efforts is considered a strength. However, the decision to rely on the containment high range radiation monitor's americium "Keep Alive" source for calibration, as discussed in Section 3 e, will be considered an unresolved item pending additional technical review by NRC. A performance based weakness was observed at Unit 1 related to the failure of the control room staff to periodically check the operability of a radiation monitoring system multi-point recorder and is described as a non-cited violation in Section 3 a.

DETAILS

1. Persons Contacted

a. Licensee

*J. Scott, General Manager, Site Chemistry
J. Albers, Manager, Radiation Protection Operations
*J. Wilson, Project Manager, Project Management Department
R. Sorensen, Chemistry/RMS Technical Services Manager
*P. Coffin, Compliance Engineer
*R. Rouse, Compliance Supervisor
*T. Murphy, RMS Supervisor
*R. Fountain, Quality Assurance (QA) & Monitoring Supervisor
*C. Gray, Unit-3 RMS Supervisor
*W. Blaxton, Unit-1 RMS Supervisor
*W. Wattson, RMS Plant System Engineer
K. Kutner, RMS/Effluents Advisor
D. Elkinton, QA Engineer
*J. Draper, Southern California Edison, Site Representative
*R. Henry, Salt River Project, Site Representative

b. NRC

*J. Sloan, Resident Inspector

(*)Denotes those individuals present at the exit interview conducted on January 3, 1992.

Additional discussions were held with other members of the licensee's staff.

2. Onsite Followup of Licensee Event Reports (LERs) and Special Reports (SRs) (92700)

Item 50-529/90-05-01 (Closed): This supplement to a 1990 SR informed the NRC that radiation effluent monitors RU-143/144 were inoperable/out of service more than 72 hours. Both RU-143/144 were out of service to allow the scheduled calibration to be performed. The calibrations actually took 312 hours to perform for various reasons. During that period, the licensee used the preplanned alternate sampling system to fulfill its safety obligation. The licensee has since devised a plan for radiation monitor (RM) calibrations and surveillance (ST) to be performed without exceeding 72 hours.

Item 50-529/90-04-01 (Closed): This supplement to a 1990 SR informed the NRC that radiation effluent monitors RU-145/146 were inoperable/out of service more than 72 hours. Both RU-145/146 were out of service to allow the scheduled calibration to be performed. The calibrations actually took 312 hours to perform due to parts replacement and modifications. During that period, the licensee used the preplanned alternate sampling system to fulfill its safety obligation. The licensee has since devised a plan that will allow RM calibrations and STs to be performed without exceeding 72 hours.

3. Gaseous Waste System: Process and Effluent Monitors (84524 & 84724)

a. Multipoint Chart Recorder 1J-SQA-RR-0029

The inspector toured Unit-1 to observe remote and local, process and effluent RMS readings to determine operability, also, to determine if the licensee replaced the multi-point chart recorder 1J-SQA-RR-0029 as referenced in an NRC Inspection Report 50-528/90-04.

During a Unit-1 control room tour on January 2, 1992, the inspector found that multipoint chart recorder 1J-SQA-RR-0029 was not printing any of its six RM data points on the chart paper. This chart recorder provides a hard copy record of radiological data for RMs RU-29, RU-31, RU-33, RU-37, RU-148, and RU-150, as discussed in Section 11.5 of the licensee's Updated Final Safety Analysis Report (UFSAR). The inspector requested that operations roll out the chart paper to determine the last time the chart recorder had printed its points. When rolled out, the chart paper revealed that the recorder had not recorded the RMS data since December 30, 1991, when the chart paper was replaced.

The inspector asked the chemistry staff and the Unit-1 control room staff who was responsible for the operation and maintenance of 1J-SQA-RR-0029. The chemistry staff was responsible for checking the operation of 1J-SQA-RR-0029 on a weekly basis. The chemistry supervisor gave the inspector a copy of the preventive maintenance (PM) work order (WO) that authorized cleaning and inspecting the recorder on December 23, 1991. After reviewing PM WO no. 508858, the inspector concluded that it was a detailed recorder PM package. The inspector reviewed Unit-1 operations requirements for assuring the recorder's operation. Appendix A of Procedure 40DP-90P05 "Control Room Data Sheet Instructions" included the 1J-SQA-RR-0029 recorder check for day and night shift. Procedure 40DP-90P05, Section 3.11, states in part that:

> "When checking control room back panels, the operators should check for normal configuration given the current plant conditions. Power, chart paper, door position, etc. should be checked. If all is normal, the operator should so indicate by placing a check mark (/) in the block. If all conditions are not normal, a note should be made at the bottom of the page in the remarks section explaining the condition. This would be considered an abnormal reading and dealt with as such.

The inspector examined control room data sheets with the 1J-SQA-RR-0029 recorder check from December 30, 1991, to January 2, 1992. The inspector found that all the recorder checks for six shifts were marked indicating all conditions were normal.

Procedure 40DP-90P05, Section 3.17, states in part that:

"Recorder charts shall be appropriately marked at the start of each day with the date and time by the individual assigned to work that area. Each shift should check each chart for synchronization with the Control Room clock, proper inking, and initial near the beginning of the shift."

The inspector found that the chart recorder 1J-SQA-RR-0029 had not been verified in accordance with the above procedure section for at least six shifts, since December 30, 1991.

This was a violation of the licensee's Technical Specification (TS) 6.8.1 which requires that written procedures shall be established, implemented, and maintained covering the activities recommended in Appendix A of Regulatory Guide (RG) 1.33, Revision 2, February 1978. RG 1.33 Appendix A.1.h, requires administrative procedures for controlling "Log Entries, Record Retention, and Procedure Review."

This violation is not being cited because the criteria specified in Section V.A. of the Enforcement Policy were satisfied (50-528/91-53-01). The licensee took prompt corrective action in restoring the chart recorder to proper operation. On January 9, 1992, the licensee issued a Unit-1 Night Order to the Unit-1 operations crew, which detailed this potential violation. The Order stressed management's expectations for correctly checking the control room data sheets, and the chart recorder. The Order was reviewed and signed by the Unit-1 operators, and the Order was issued to the Units 2 and 3 operations staff. The Order stated that a Condition Report Disposition Request (CRDR) No. 1-2-009 was written to investigate this chart recorder problem. Additionally, the inspector observed a Unit-1 reactor operator adequately perform the multipoint recorder check on January 3, 1992. The inspector had no further concerns in this matter.

b. <u>Incident Investigation Report (IIR) Radiation Monitoring Systems (RMS)</u>

The inspector examined the results of IIR 3-1-90-65, completed December 12, 1991, in which the licensee assessed the extent of RMS licensing document discrepancies. During a 10 CFR 50.59 safety evaluation on revising the RMS alarm setpoint procedure, the licensee found that several RMS design basis documents were incorrect (i.e. TS, UFSAR, Design Criteria Manual & RMS Description). Since this concern had safety implications, the licensee's Plant Review Group (PRG) tasked nuclear instrumentation & controls engineering (NICE) to lead an investigation. This NICE IIR consolidated several different licensee reports on RMS problems such as: Problem Resolution Sheet (PRS), Quality Deficiency Report (QDR), CRDR, Engineering Evaluation Report (EER) and a vendor report. This IIR 3-1-90-65 raised the following questions about the RMS:

* What was the calculational basis of the setpoints for the RMS?

* Why do the UFSAR, RMS description, design criteria and other relevant plant documents not correctly reflect the present field equipment configuration?

* What administrative procedural faults permitted the RMS documents to become inaccurate?

The licensee's IIR found:

* The RMS design basis calculations 13C-SQ001 were superseded.

* Calculations for most RMS setpoints were not readily available.

* RMS setpoints were different from one design document to another.

* Design changes were implemented without updating RMS documents.

* RMS temporary modifications (T-mod) were in place for too long.

* Adherence to document control guidelines were not mandated.

* Situations existed that allowed RMS changes without updating design record.

* T-mods were installed without NICE concurrence.

* RMS changes were not evaluated for operation prior to installation.

* RMS field configuration and design documents were deficient and inaccurate.

The licensee's IIR conclusions and corrective actions were as follows:

* There was no adequate source or justification for design basis calculations 13C-SQ001 and RMS setpoints. The superseded setpoint design criteria and basis for RMS will be re-established during the Setpoint and Design Basis Reconstitution program, which will be complete in June 1992.

* Stricter administrative controls guidelines will be established to ensure that RMS design changes are updated promptly and accurately incorporated into the appropriate licensing documents.

* RMS field installed configurations will be compared to the TSs, UFSAR and other licensing design documents.

The inspector concluded that the IIR process provided an integrated approach to understanding and resolving RMS problems. The inspector

reviewed the operability status of the RMS, and found it to be as required by the TS and UFSAR. The inspector reviewed the current RMS setpoints as described in Section 3(c) of this report. The RMS adequately performed its designed safety objectives. The inspector had no further concerns in this matter.

c. RMS Setpoints Basis vs Regulatory Guide 1.105

The inspector examined the current status of the RMS setpoint program with respect to operational safety. In Section (b) of this report, NICE committed to re-establishing new RMS design basis setpoints by June 1992. The licensee's setpoint program was committed to Regulatory Guide (RG) 1.105, "Instrument Setpoints," November 1976, Revision 1, by the UFSAR Chapter 1.8. The licensee's Engineering Evaluation Request (EER) No. 90-SQ-100, dated November 28, 1990, re-evaluated the applicability of RG 1.105 to safety related RMS setpoints in regards to calculating instrument loop uncertainty errors and setpoint errors. The EER also examined the basis of the current operating RMS setpoints. The inspector examined the EER's findings and discussed them with the RMS engineer who dispositioned EER 90-SQ-100. The inspector had discussions with NICE who will be performing the setpoint reconstitution and writing TS interpretations on setpoints/limits. The EER 90-SQ-100 found that the original RMS setpoint calculation basis were superseded, but it did not mean that the current RMS setpoints used for operations had no documented basis. Additionally, the EER clarified that RMs RU-30, RU-31, RU-37, RU-38, and RU-145 were safety related. The EER stated that RG 1.105 was not applicable, and RMS setpoints were in accordance with the Offsite Dose Calculation Manual (ODCM). The EER further justified the view that each TS setpoint contained a high degree of conservatism. This view was consistent with the licensee's TS Interpretation (TSI)# 201 dated June 12, 1987, which stated in part:

> "Absolute values listed in the TS are assumed to be limits which are not exceeded in the safety analysis. To maintain validity of the safety analysis, these values must not be exceeded."

However, the inspector noted that TSI# 201 was superseded by TSI 13-07-00 effective September 10, 1991, which states in part:

> "Values in the TS were derived using the criteria of RG 1.105 and such conservatism has already been applied and therefore further "inaccuracies" or "tolerance" cannot be applied since the margins used in the safety analysis would be compromised."

The inspector discussed with NICE and the RMS engineer the contradictory TSI positions with regard to RG 1.105. The RMS engineer issued a TS interpretation change request on January 3, 1992, to clear up the TSI problem. The RMS engineer stated that RMS setpoints listed in the TS contain a sufficient safety margin, and the TSI needs to reflect that items specifically called out as "setpoints" and be treated as such. NICE held a meeting on

January 3, 1991, and gave a memorandum to the inspector on the RMS setpoint program, its applicability to RG 1.105, EER 90-SQ-100, and the TSIs. NICE committed that reevaluation of the RMS setpoints will be completed by June 1992. NICE decided that RG 1.105 did not strictly apply to RMS setpoints. However, NICE concluded that under the setpoint reconstitution program PVNGS was required to comply with RG 1.105, to be consistent with industry standard ISA-S67.04-1988, "Setpoints for Nuclear Safety-Related Instrumentation."

The inspector determined that the issues were:

* whether or not the original RMS setpoints, 13-SQ001, during the design basis, had an adequate safety margin calculated into the setpoints to assure that the safety limit parameter would not be exceeded due to instrument loop and RMS inaccuracies;

* when the original 13-SQ001 design basis calculations were superseded, did subsequent setpoint determinations re-establish a safety margin.

The inspector concluded that based on the licensee's efforts on the Setpoint Reconstitution program and the EER:

* The licensee originally complied with the intent of RG 1.105 to have adequate safety margins in the RMS instrument loops.

* Although the licensee superseded the original RMS design basis calculations (13-SQ001), the current RMS setpoints are still consistent with 13-SQ001, and the ODCM. The RMS Setpoint Reconstitution will reassure that the intent of RG 1.105 Revision 1 is met.

The inspector examined the licensee's current "RMS Effluents Monitor Setpoint Calculations for 1991," dated January 30, 1991. These setpoint calculations were based on 1% failed fuel mix as specified by procedure 74RM-9EF42. The licensee determined that the 1991 setpoints would be unchanged from the 1990 setpoints. The inspector verified that alarm/trip setpoints for the RMS were maintained in accordance with ODCM and applicable procedures. The inspector had no further concerns with this matter.

d. <u>RMS Light Emitting Diodes (LEDs) for Source Checks</u>

The inspector examined the process and effluent RMS source check program to determine if it was in compliance with TS Table 4.3-8. On April 30, 1991, the licensee submitted a proposed change to TS Table 4.3-8, "Radioactive Gaseous Effluent Monitoring Instrumentation Surveillance Requirements." On September 26, 1991, the NRC's Office Nuclear Reactor Regulation (NRR) approved the proposed TS change as Amendments Nos. 56, 43,and 29 for Units 1, 2, and 3 respectively. The TS Amendments specifically allowed the licensee to use LEDs as source checks for noble gas activity monitors RMs RU-12, RU-141, RU-143, and RU-145. The licensee's TS

1.33 defined a Source Check as "a qualitative assessment of channel response when the sensor is exposed to a source of increased radioactivity." TS Table 4.8.3-8, "Table Notations," number (7) was added, and it states that, "LED may be utilized as the check source in lieu of a source of increased radioactivity." A major consideration in the NRC's decision to allow this TS change was the licensee's EER 90-SQ-094 which stated that the RMs had no credible failure that could be detected by a radioactive source that a LED could not detect also.

The inspector examined the licensee's procedures 74ST9SQ04 and 74ST9SQ06 for performing source checks on the RMs during the interim period prior to approval of the TS Amendment. Licensee Event Report (LER) No. 90-012-00, dated December 26, 1990, addressed the licensee's use of LEDs contrary to the TS 1.33 definition. The LED issue in LER 90-012-00 ,also, addressed in NRC Region V Inspection Report 50-528/91-13, stated that the licensee would submit a TS change proposal to the NRC by April 30, 1991. The LER explained that procedural controls were in place for the RMs to be source checked using a radioactive source. The inspector confirmed that the procedures had provisions to use radioactive sources as alternate check sources.

The inspector had no further concern in this matter.

e. RMS Calibrations and Detector/Laboratory Comparisons

The inspector verified that surveillance requirements for the RMS were being maintained and implemented by the methods allowed by TS Table 3.3-6, TS Table 4.3-8, NUREG 0737, Table II.F.1-3, and UFSAR Chapter 11.5.2.

Process and Effluents RMS

The inspector discussed RMS calibration and detector/laboratory comparison programs with the RMS engineer, the RMS/effluents supervisor, and a RMS technical advisor. The licensee routinely performed cross checks between effluent lab samples and RMS readings as additional verification of RMS accuracy. If there was a 30% variance between an effluent lab sample and a RMS reading, the RMS supervisor was notified for advice. The licensee did not write a procedure for this comparison process, nor was it specifically committed to in the UFSAR. However, the effluents comparison process routinely assured that permit release rates and RMS setpoints were not exceeded.

The licensee's UFSAR Chapter 11.5.2 does not require isotopic calibrations of the RMS, only a single point calibration to confirm detector sensitivity. Full isotopic calibrations were performed at the factory, and the factory provided field calibration sources and reference decay curves. The UFSAR states that the RMS detector geometries cannot be altered, therefore, subsequent calibrations were based on known correlations between the detector response and

field calibration standards. The inspector pointed out that TS Table 4.3-8 Notation (3) requires the initial channel calibration be performed using one or more certified National Institute for Standards and Technology (NIST) radiation sources or factory obtained standards that were traceable to NIST. The licensee representative stated that the initial calibration on the RMS was performed at the factory, and that only the last sentence in TS Notation (3) applied. That last sentence requires subsequent channel calibrations to use sources that were related to the initial calibration. The licensee gave a copy of EER 89-SQ-157, completed June 14, 1991, to the inspector to examine. EER 89-SQ-157 was an energy response test of eight RMs using three different beta radiation sources (Tc-99, Cl-36 & Sr-90) that were similar to what was used at the factory initial calibration. The test objective was to determine if primary "In Situ" calibrations were needed on RMS effluent and process monitors as suggested by EER 86-SQ-030. The results of the test suggested no "In Situ " calibration was necessary, because none of the RMs tested had a response which differed by more than 15% from the factory calibrations. The inspector concluded that the process and effluent RMS response was acceptable.

Containment High Range Area Monitors

The inspector examined the area radiation monitoring instrumentation program. The calibration and test requirements for this part of the RMS were in TS Table 3.3-6 and UFSAR Chapter 11.5. The inspector examined the calibration methods for Containment High Range Monitors (CHRMs) RU-148 and RU-149. The licensee's UFSAR Chapters 11.5 and 18.II.F.1.3 committed them to additional requirements for the Containment High Range Monitors found in NUREG-0737, Table II.F.1-3. The licensee's TS required that CHRMs RU-148 and RU-149 receive channel calibrations every 18 months. A channel calibration is defined in TS 1.4 as:

> "The adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The channel calibration shall encompass the entire channel including the sensor and alarm and/or trip functions..."

NUREG-0737, Table II.F.1-3 requires that CHRMs have in situ calibrations for at least one decade below 10 Roentgen/hour (R/hr) by a calibrated radiation source. Additionally, the NUREG states the original laboratory calibration is not an acceptable position due to the possible differences after installation.

The licensee's representatives gave the inspector the following documents for review:

* Procedure 36ST-9SQ08, "Radiation Monitoring Calibration Test for New Scope Area Monitors," and Procedure Change Notice (PCN) No. 3, dated October 21, 1991.

* EER 90-SQ-107, dated December 20, 1990, on CHRMs RU-148 and RU-149 evaluated the licensee position on complying with NUREG-0737. The EER suggested that in situ tests be performed, since the original installation at Palo Verde did not include the test.

* Instruction Change Request (ICR) 18057, was initiated December 20, 1990, and completed October 23, 1991. The ICR allowed the CHRMs to use internally mounted Am-241 sources for meeting the intent of NUREG 0737 Table II.F.1-3, with regard to the in situ calibrations and a radiation source at least 1 decade below 10 R/hr.

The inspector's review centered around two questions related specifically to the licensee's CHRM design:

* Was the Am-241 internal radiation source equivalent to the NUREG-0737 calibrated radiation source at least one decade below 10 R/hr, and not an electronic calibration?

* Was the NUREG-0737 in situ calibration requirement satisfied by the CHRM internal test Am-241 radiation source?

The inspector reviewed vendor documents on the CHRM type ionization chambers, factory calibrations, and the internal Am-241 source. Additionally, the inspector reviewed licensee's CHRM calibration data, calibration procedure, and the test results of EER-90-SQ-107. According to the vendor data, the operating characteristics of the Am-241 internal test source was equivalent to a 1 to 5 R/hr source. Each Am-241 source generates a continuous current of about 1E-11 to 5E-11 amps at the time of primary calibration, which was equivalent to gamma response sensitivity in units of amps/R/hr. The licensee's EER-90-SQ-107 tested the CHRMs (RU-148/149) for Units-1, 2, & 3 by comparing the internal radiation source results to the primary calibration. The internal source test results for all CHRMs were within ±20% of the primary calibration data. The licensee's ICR No. 18057, allowed calibration procedure 36ST-9SQ08 for the CHRM to have an acceptance criteria of ±30%. The licensee used this test data to justify why in situ calibrations, as suggested by the EER were unnecessary.

The inspector noted that vendor documents referred to the internal radiation source test method as an indication of adequate electronic calibration, and that reference may have lead to confusion on whether PVNGS met the NUREG-0737 requirements of "In situ" calibration by electronic signal substitution for all ranges above 10 R/hr." The licensee's procedure 36ST-9SQ08, clearly used electronic signals for all ranges above 10 R/hr, during CHRM calibrations, and the internal radiation source for the range one decade below 10 R/hr.

As previously stated in this report, the licensee's UFSAR requires only a single point calibration with the factory provided source and reference curve for verifying RMS sensitivity.

On the question of the licensee performing in situ calibrations on the CHRMs, again, the UFSAR stated that the RMS detector geometries cannot be altered, therefore, subsequent calibrations were based on known correlations between the detector response and field calibrations standards. Additionally, the licensee's EER 90-SQ-107 and ICR-18057 validated their position with regard to in situ calibrations on CHRMs and NUREG-0737.

The technical merits of the licensee's method for meeting NUREG-0737 requirements using the internal Am-241 source was discussed with NRR and will be considered an unresolved item pending further review (50-528/91-53-01).

An unresolved item is a matter about which more information is required to ascertain whether it is an acceptable item, a deviation, or a violation.

The licensee's RMS programs appeared to meet the safety objectives of the TS, UFSAR Chapter 11.5, and the ODCM. One non-cited violation, and one unresolved item were identified; no deviations were identified.

4. Exit Interview

The inspector met with the individuals denoted in Section 1 at the conclusion of the inspection on January 3, 1992. The scope and findings of the inspection were summarized. The licensee was informed of the non-cited violation discussed in Section 3(a). The licensee acknowledged the inspector's observation.

U.S. NUCLEAR REGULATORY COMMISSION

Revision 2*
February 1986

REGULATORY GUIDE

OFFICE OF NUCLEAR REGULATORY RESEARCH

REGULATORY GUIDE 1.105
(Task IC 010-5)

INSTRUMENT SETPOINTS FOR SAFETY-RELATED SYSTEMS

A. INTRODUCTION

Criterion 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires, among other things, that instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables and systems within prescribed operating ranges.

Criterion 20, "Protection System Functions," of Appendix A to 10 CFR Part 50 requires, among other things, that the protection system be designed to initiate operation of appropriate systems to ensure that specified acceptable fuel design limits are not exceeded.

Paragraph (c)(1)(ii)(A) of § 50.36, "Technical Specifications," of 10 CFR Part 50 requires that, where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting be so chosen that automatic protective action will correct the most severe abnormal situation anticipated without exceeding a safety limit. It also requires the licensee to notify the NRC of any automatic safety system malfunctions, to review the matter, and to record the results of the review. Setpoints that exceed technical specification limits are considered a malfunction of an automatic safety system.

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations for ensuring that instrument setpoints are initially within and remain within the technical specification limits.

The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

*The substantial number of changes in this revision has made it impractical to indicate the changes with lines in the margin.

Any information collection activities mentioned in this regulatory guide are contained as requirements in 10 CFR Part 50, which provides the regulatory basis for this guide. The information collection requirements in 10 CFR Part 50 have been cleared under OMB Clearance No. 3150-0011.

B. DISCUSSION

Revision 1 to Regulatory Guide 1.105, "Instrument Setpoints," was published in November 1976 in response to the large number of reported instances in which instrument setpoints in safety-related systems drifted outside the limits specified in the technical specifications. Using the method described in Revision 1 to Regulatory Guide 1.105 and additional criteria on establishing and maintaining setpoints, Subcommittee SP67.04, Setpoints for Safety-Related Instruments in Nuclear Power Plants, under the Nuclear Power Plant Standards Committee of the Instrument Society of America (ISA) has developed a standard containing minimum requirements to be used for establishing and maintaining setpoints of individual instrument channels in safety-related systems. This standard is ISA-S67.04-1982, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants."**

Some key terms used throughout ISA-S67.04-1982 are not defined or have unclear applications. For convenience, the following information is provided: (1) the definition of the term "safety limit" is contained in § 50.36 of 10 CFR Part 50, (2) the term "allowable value" as used in the standard is consistent with the usage in the bases sections of the Standard Technical Specification (STS),*** (3) the term "upper setpoint

**Copies are available from the Instrument Society of America, P.O. Box 12277, Research Triangle Park, North Carolina 27709.

***NUREG-0103, Revision 4, "Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors"; NUREG-0123, Revision 3, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/5)"; NUREG-0212, Revision 2, "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors"; and NUREG-0452, Revision 4, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors." Copies of NUREG-series documents may be purchased from the Superintendent of Documents, U.S. Government Printing Office, Post Office Box 37082, Washington, DC 20013-7082.

limit" as used in Figure 1 of the standard is the same as "trip setpoints" as used in the aforementioned STSs in that drift above the "upper setpoint limit" (standard) or "trip setpoint" (STSs) requires readjustment.

Paragraph 4.3 of the standard specifies the methods for combining uncertainties in determining a trip setpoint and its allowable values. Typically, the NRC staff has accepted 95% as a probability limit for errors. That is, of the observed distribution of values for a particular error component in the empirical data base, 95% of the data points will be bounded by the value selected. If the data base follows a normal distribution, this corresponds to an error distribution approximately equal to a "two sigma" value.

Section 6 requires that "software qualification" be documented. Although there is no generally accepted definition in the nuclear industry for software qualification; the industry has used ANSI/IEEE-ANS-7-4.3.2-1982, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," for verification and validation of computer software used in safety-related systems. Regulatory Guide 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants," endorses this standard.

Some of the considerations in documenting setpoint drift are (1) the degree of redundancy of the channels for which the allowable limits have been exceeded, (2) the type of instrument, including the instrument's designed accuracy, function, and plant identification number, (3) the allowable value in the technical specifications, (4) the "as left" setpoint from prior surveillance, (5) the measured setpoint, (6) the amount of adjustment in the reported occurrence and the current "as left" setpoint, and (7) the history of previous testing and the amount of any drift and adjustment in previous testing.

C. REGULATORY POSITION

ISA-S67.04-1982, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," establishes requirements acceptable to the NRC staff for ensuring that instrument setpoints in safety-related systems are initially within and remain within the technical specification limits. The last section of ISA-S67.04-1982 lists additional standards that are referenced in other sections of the standard. Those referenced standards not endorsed by a regulatory guide (or incorporated into the regulations) also contain valuable information and, if used, should be used in a manner consistent with current regulations.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the NRC staff's plans for using this regulatory guide.

Except in those cases in which the applicant or licensee proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the methods described in this guide will be used by the NRC staff in the evaluation of instrument setpoints for safety-related systems with respect to the technical specification limits for the following nuclear power plants:

1. Plants for which the construction permit is issued after February 1986.

2. Plants for which the operating license application is docketed 6 months or more after February 1986.

3. Plants for which the applicant or licensee voluntarily commits to the provisions of this guide.

VALUE/IMPACT STATEMENT

1. BACKGROUND

The most common cause of a setpoint in a safety-related system being out of compliance with plant technical specifications has been the failure to allow for a sufficient margin to account for instrument inaccuracies, expected environmental drift, and minor calibration variations. For example, in some cases, the trip setpoint selected was numerically equal to the allowable value and stated as an "absolute value," thus leaving no apparent margin for drift. In other cases, the trip setpoint was so close to the upper or lower limit of the range of the instrument that instrument drift placed the setpoint beyond the range of the instrument, thus nullifying the trip function. Other general causes for a setpoint being out of conformity with the technical specifications have been instrument design inadequacies and questionable calibration procedures.

Revision 1 to Regulatory Guide 1.105, "Instrument Setpoints," was issued in November 1976 in response to the large number of instances reported in Licensee Event Reports (LERs) of setpoints drifting outside the limits specified in the technical specifications. Revision 1 provided general guidance for (1) specifying setpoints (by considering instrument drift, accuracy, and range) and (2) having a securing device for the setpoint adjustment mechanism.

The method described in Revision 1 to Regulatory Guide 1.105 has been incorporated into an Instrument Society of America Standard, ISA-S67.04-1982, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants." Revision 2 to Regulatory Guide 1.105 was developed to use the guidance of ISA-S67.04-1982. This revision provides more specific guidance on establishing and maintaining setpoints in response to the needs that were apparent from (1) a continuing large number of reportable occurrences and (2) the licensing review of methodology for specifying allowable values and trip setpoints.

2. VALUE/IMPACT ASSESSMENT

2.1 General

ISA-S67.04-1982 is considered state-of-the-art methodology for specifying and reviewing technical specifications on allowable values and trip setpoints, and members of the industry have incorporated this standard into their internal procedures. Further, paragraphs 50.73(a) and (b) of 10 CFR Part 50 define when an LER is required and what is to be included in an LER, respectively.

2.2 Value

The value to NRC operations and industry is that there would be (1) a systematic method for specifying and reviewing technical specifications on allowable values and trip setpoints, (2) more sophisticated methods for specifying technical specifications, (3) a reduction in setpoint readjustments, (4) less chance for unwarranted reactor shutdown, and (5) fewer LERs and other reportable occurrences from the allowable limits of setpoints being exceeded.

2.3 Impact

The impact would be minimal as ISA-S67.04-1982 represents current industry practice that has been codified in a national consensus standard.

U.S. NUCLEAR REGULATORY COMMISSION

Revision 1
November 1976

REGULATORY GUIDE

OFFICE OF STANDARDS DEVELOPMENT

see proposed Rev. 2

REGULATORY GUIDE 1.105

INSTRUMENT SETPOINTS

A. INTRODUCTION

Criterion 13, "Instrumentation and Control," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires, among other things, that instrumentation be provided to monitor variables and systems and that controls be provided to maintain these variables and systems within prescribed operating ranges.

Paragraph (c)(1)(ii)(A) of §50.36, "Technical Specifications," of 10 CFR Part 50 requires that, where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting be so chosen that automatic protective action will correct the most severe abnormal situation anticipated before a safety limit is exceeded.

This guide describes a method acceptable to the NRC staff for complying with the Commission's regulations with regard to ensuring that the instrument setpoints in systems important to safety initially are within and remain within the specified limits. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

Operating experience has shown that there is need for guidance in the selection of required instrument accuracy and the settings that are used to initiate automatic protective actions and alarms.

Abnormal Occurrence Reports submitted by operating utilities between January 1972 and June 1973 record the most frequent abnormal occurrence as the drift of the protective instrument setpoint outside the limits specified in the technical specifications.

Protective instruments and alarms in nuclear power plants are provided with adjustable setpoints where specific actions are either automatically initiated, prohibited, or alarmed. For example, pressure sensors typically are installed on main steam lines to measure steam pressure. These sensors initiate corrective action if the steam pressure decreases to the predetermined and preset value that would result, for example, from a steam line break. Setpoints (e.g., pressure, differential pressure, flow, level, temperature, power, radiation level, time delay) correspond to certain provisions of technical specifications that have been incorporated into the operating license by the Commission.

The single most prevalent reason for the drift of a measured parameter out of compliance with a technical specification is the selection of a setpoint that does not allow a sufficient margin between the setpoint and the technical specification limit to account for inherent instrument inaccuracy, expected vibration, and minor calibration variations. In some cases, the setpoint selected was numerically equal to the technical specification limit and stated as an absolute value, thus leaving no apparent margin for error. In other cases, the setpoint was so close to the upper or lower limit of the instrument's range that the instrument drift placed the setpoint beyond the instrument's range, thus nullifying the trip function. Other causes for drift of a parameter out of conformity with a technical specification have been instrumentation design inadequacies and questionable calibration procedures.

The following terms are listed with the definitions used in this guide:

1. **Instrument accuracy**—the degree to which an indicated value conforms to an accepted standard value or a true value.

* Lines indicate substantive changes from previous issue.

2. **Drift**—a change in the input-output relationship of an instrument over a period of time.

3. **Margin**—the difference between a limiting condition and an operating condition.

4. **Range**—the region within which a quantity is measured, received, or transmitted.

5. **Safety limit**—a limit on an important process variable that is necessary to reasonably protect the integrity of physical barriers that guard against uncontrolled release of radioactivity.

6. **Setpoint**—a predetermined level at which a bistable device changes state to indicate that the quantity under surveillance has reached the selected value.

7. **Span**—the algebraic difference between the upper and lower limits of the range.

8. **Technical specification limit**—the limit prescribed as a license condition on an important process variable for safe operation.

9. **Systems important to safety**—those systems that are necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR Part 100, "Reactor Site Criteria."

C. REGULATORY POSITION

The following are applicable to instruments in systems important to safety:

1. The setpoints should be established with sufficient margin between the technical specification limits for the process variable and the nominal trip setpoints to allow for (a) the inaccuracy of the instrument, (b) uncertainties in the calibration, and (c) the instrument drift that could occur during the interval between calibrations.

2. All setpoints should be established in that portion of the instrument span which ensures that the accuracy, as required by regulatory position 4 below, is maintained. Instruments should be calibrated so as to ensure the required accuracy at the setpoint.

3. The range selected for the instrumentation should encompass the expected operating range of the process variable being monitored to the extent that saturation does not negate the required action of the instrument.

4. The accuracy of all setpoints should be equal to or better than the accuracy assumed in the safety analysis, which considers the ambient temperature changes, vibration, and other environmental conditions. The instruments should not anneal, stress relieve, or work harden under design conditions to the extent that they will not maintain the required accuracy. Design verification of these instruments should be demonstrated as part of the instrument qualification program recommended in Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants."

5. Instruments should have a securing device on the setpoint adjustment mechanism unless it can be demonstrated by analysis or test that such devices will not aid in maintaining the required setpoint accuracy and minimizing setpoint changes. The securing device should be designed so that it can be secured or released without altering the setpoint and should be under administrative control.

6. The assumptions used in selecting the setpoint values in regulatory position 1 and the minimum margin with respect to the limiting safety system settings, setpoint rate of deviation (drift rate), and the relationship of drift rate to testing interval (if any) should be documented.

D. IMPLEMENTATION

The purpose of this section is to provide information to applicants and licensees regarding the staff's plans for utilizing this regulatory guide.

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the method described herein will be used in the evaluation of submittals in connection with construction permit applications docketed after December 15, 1976.

If an applicant wishes to use this regulatory guide in developing submittals for applications docketed on or before December 15, 1976, the pertinent portions of the application will be evaluated on the basis of this guide.

TABLE II.F.1-3

CONTAINMENT HIGH-RANGE RADIATION MONITOR

REQUIREMENT	-	The capability to detect and measure the radiation level within the reactor containment during and following an accident.
RANGE	-	1 rad/hr to 10^8 rads/hr (beta and gamma) or alternatively 1 R/hr to 10^7 R/hr (gamma only).
RESPONSE	-	60 keV to 3 MeV photons, with linear energy response ± 20%) for photons of 0.1 MeV to 3 MeV. Instruments must be accurate enough to provide usable information.
REDUNDANT	-	A minimum of two physically separated monitors (i.e., monitoring widely separated spaces within containment).
DESIGN AND QUALIFICATION	-	Category 1 instruments as described in Appendix A, except as listed below.
SPECIAL CALIBRATION		In situ calibration by electronic signal substitution is acceptable for all range decades above 10 R/hr. In situ calibration for at least one decade below 10 R/hr shall be by means of calibrated radiation source. The original laboratory calibration is not an acceptable position due to the possible differences after in situ installation. For high-range calibration, no adequate sources exist, so an alternate was provided.
SPECIAL ENVIRONMENTAL QUALIFICATIONS	-	Calibrate and type-test representative specimens of detectors at sufficient points to demonstrate linearity through all scales up to 10^6 R/hr. Prior to initial use, certify calibration of each detector for at least one point per decade of range between 1 R/hr and 10^3 R/hr.