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David L. Meyer, Chief
Rules and Directives Branch
Nuclear Regulatory Commission
Washington DC, 20555-0001

Attention: Rulemakings and Adjucations Staff

Mr. Meyer:

Enclosed are my comments regarding the proposed modification of NRC requirements for Post Accident Sampling Systems (PASS). Public comment on the proposed changes was requested in Federal Register Volume 64, Number 226, pages 66213-66214.

Although I am an engineer in the employ of Arizona Public Service Company, which owns and operates the Palo Verde Nuclear Generating Station, these are my own personal opinions and do not necessarily reflect the views of the company or its management. Since my opinions will be relevant to the subsequent development of revised PASS criteria at Palo Verde, I have been allowed the use of company time and resources to prepare these remarks. Any questions about my comments may therefore be directed to me at my work telephone (623)393-6980 or the business address below.

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EVALUATION of the PROPOSAL to ELIMINATE the POST-ACCIDENT SAMPLING SYSTEM (PASS)

Introduction

A Post-Accident Sampling System (PASS) consists of the sample connections, tubing/piping, and instrumentation needed to perform chemical and radiochemical analyses of post-accident fluids during a reactor accident. Following a reactor accident, PASS is used to provide utility and government staff with the information required to recommend actions for the protection of public health and safety and also to verify that safety systems are operating as designed. Such systems were required to be installed at all US commercial power reactors following the accident at Three Mile Island Unit II in March 1979. Due to number of factors, such systems have proved to be difficult and expensive for the nuclear power industry to test and maintain. Consequently, the utilities through their Nuclear System Supply System (NSSS) Owner's Groups have petitioned the NRC to eliminate the regulatory requirement for PASS. The NRC has concurred with the industry position and developed a proposed rule making, which has been approved by the Advisory Committee on Reactor Safeguards (ACRS).

The proposal to eliminate PASS requirements altogether is fundamentally based on the contention that the essential information provided by PASS -- that used for the purpose of accident mitigation and protective action decision making -- is available with the required accuracy from other sources. This is not true for the Combustion Engineering (CE) System 80 reactor plant. While this conclusion may also correct for other pressurized water reactor designs, the following discussion was preparing considering only the CE System 80 design and the core damage assessment methods presented in CE Owner's Group Task 467 (Reference 11). This position paper has been written to show that the justification offered for the proposed rulemaking is not technically sound and that the safety consequences of PASS elimination have therefore not been fully considered by the NRC and ACRS.

The Purpose of PASS

PASS allows the collection and analysis of high specific activity samples of post-accident fluids without over-exposure of the chemistry technician. The system is intended to aid in the mitigation of and the emergency response to reactor transients by providing information related to the assessment of core damage, actual and projected offsite doses, and also the effectiveness of safety systems.

The PASS results and the analyses derived from them may factor into the overall emergency response effort through a variety of overlapping mechanisms:

- The emergency is classified in part by assessment of the status of the three principal fission product barriers: fuel clad, reactor coolant system piping, and the containment. Coolant activity and more advanced core damage assessments are used to evaluate the clad barrier. Event classification is then used in part to recommend offsite protective actions.
- The emergency is also classified in part by projected doses and dose rates at the site boundary. Since PASS results may be used to perform offsite dose assessment, they may also affect event classification in this way as well.

- In order to insure timeliness, protective actions may be made based on the presence of large amounts of activity in containment available for release. Use of plant status in protective action decision making is discussed in IE Information Notice 83-28.

- Protective actions may also be based on the dose consequences of actual or potential releases of radioactive materials. Both of these types of calculations can be performed using PASS data.

- PASS data can be used to verify in-plant instrument and monitor readings and to perform independent verification of many safety processes including emergency core cooling, containment spray (fission product removal), reactivity control, and combustible gas control, containment sump iodine retention. This information will either confirm that the equipment is performing as designed or alert plant staff to jeopardized safety functions. Emergency response staff resources and priorities may then be adjusted accordingly.

The regulatory requirements for the utility to classify the emergency, assess the potential offsite consequences of the event, and make protective action recommendations to secure the public health are contained in 10 CFR 50.47 and 10 CFR 50 Appendix E.

Core Damage Assessment

Per CE Task 467, the technical objective of core damage assessment is classify the degree of fuel damage into one of the three damage groups recommended by the Rogovin Report: cladding failure, fuel overheat, and pellet melt. The procedure goes further into distinguishing between minor (less than 10 % of fuel pins affected), intermediate (10%-50%), and major (greater than 50%) categories within each group. The categories are numbered sequentially with category 1 being "no damage" (< 1% clad damage) and then increasing in severity for a total of 10 damage categories overall.

CE Task 467 identifies four methods for performing core damage assessment:

- Core Exit Thermocouples (CETs)
- Containment Area Radiation Monitors
- Containment Hydrogen
- Radiochemical Analysis of Post-Accident Fluids

Since the last method requires the use of PASS, only the first three methods would be available for core damage assessment if PASS were not available.

The CE System 80 design contains about 61 thermocouples spread across the entire core cross-section located approximately 15" above the top of the active fuel. The objective of the CET core damage assessment method is to determine the fraction of fuel pins that have damaged cladding as a result of the ballooning and burst/crack mechanism. The susceptibility of a fuel pin to fail by this mechanism depends on the temperature of the clad and the differential pressure between the interior and exterior of the pin. The method starts by estimating the clad temperature where the balloon/burst mechanism is possible as a coarse function of the reactor coolant pressure at the time of core uncovery. Then the analyst records the maximum core exit thermocouple temperature. Assuming that the radial temperature distribution across the core cross section varies as the does power density during normal operation, a graph is provided to indicate what fraction of the fuel pins in the core have temperatures higher than the expected clad failure temperature.

This method when used alone has several limitations with respect to accurate core damage assessment:

- The method cannot identify whether fuel overheat or pellet melting have occurred.
- The method assumes that the radial temperature profile is similar to the radial distribution of neutron flux (power); therefore, it cannot be used to quantify core damage as a result of loose parts or local obstruction of coolant flow, such as occurred at the Fermi I reactor.
- The method assumes that CET temperature is representative of fuel clad temperature. Since this is only possible when there is a flow of steam from the core to the CETs, the method is useless following degraded coolant flow. The reduction in heat transfer results in an adiabatic core heatup where the CET temperature becomes "uncoupled" from the fuel temperature.

The Containment Area Radiation Monitor core damage assessment method does not rely on PASS as well. These high-range ion chambers are used to detect gross gamma radiation levels from activity in the containment. CE Task 467 provides a graph with a family of curves representing the expected monitor response for 10% clad failure, 50% clad failure, 10% fuel overheat, and 50% overheat as function of time from shutdown. The analyst may scale the actual monitor reading based on the power history before the event. Thus, the damage category can be determined knowing only the monitor reading, power history, and the time after shutdown.

This method also has limitations:

- The method cannot detect if significant fuel melt has occurred.
- The CE Task 467 curves were developing assuming that 100% of the noble gases, 25% of the iodine, and 1% of the particulates in the fraction of damaged fuel was available for release. This source term assumption was taken from the Regulatory Guides. They were intended to be conservative with respect to offsite dose calculations and do not represent any sort of best engineering judgment as to what the actual nuclide mix might be. Since the best available data suggest that large fractions of radionuclides will be retained in the sump liquid, the fraction of fuel damage predicted with the CTMT radiation monitors may be severely underestimated.
- The response of a gross gamma detector is highly dependent on the assumed distribution of radionuclides. Since the nuclide mix varies greatly from one accident scenario to the next, the actual monitor response may vary by orders of magnitude from that predicted by the calculation.

Core damage can be estimated based on knowledge of containment hydrogen concentration that can be obtained by in-line hydrogen analyzers and therefore does not require PASS. As described in CE Task 467, the hydrogen analyzer is used to determine the total amount of hydrogen in containment. This value is corrected for hydrogen contributions from oxidation of aluminum and zinc in containment, radiolysis of water, and hydrogen gas normally dissolved in the reactor coolant. The remaining hydrogen is attributed to core-wide oxidation of the fuel cladding. The Task provides a graph that can be used to estimate the fraction of fuel pins that have been embrittled from the zircalloy/steam oxidation reaction, which proceeds significantly at high (overheat) cladding temperatures.

The limitations of this method are:

- The method is only valuable for detecting fuel overheat. The hydrogen monitors, which are designed primarily to detect potentially flammable concentrations of hydrogen in containment, are ranged such that the instrument will not come on-scale until fuel overheat has occurred (damage category 4 or higher). It provides no information about pellet melt.

- The method has a large degree of uncertainty due to the fact that is assumes that zircalloy/steam reaction is never rate limited by an absence of steam. Thus, the method is least accurate in the cases where fuel overheat is expected (i.e., complete loss or degraded coolant flow).

- There is additional uncertainty associated with the ability to analyze a representative hydrogen sample. First, it is possible to trap a significant fraction of hydrogen in the reactor vessel upper head. Second, because hydrogen gas is so much lighter than air, it is possible that a large fraction of hydrogen gas can be trapped in the upper containment dome, even if the containment atmosphere is well mixed due to the operation of containment spray. Note that the suction for the hydrogen analyzers is taken below the site grade level.

While there are also limitations to performing core damage assessment based on radiochemical analysis of the coolant, containment sump, and containment atmosphere, it is by far -- in most accident scenarios -- the most accurate of the four methods. More importantly, it is the only method that can be valid over the whole range of core damage and is the only method capable of identifying whether significant pellet melting has occurred. The accuracy of the assessment depends on how closely the real event matches the assumptions used to develop each of the methods. The basis document, CE Task 467, indicates that the analyst has the most confidence in the damage classification when all four methods are used together and they indicate consistently or when the differences can be reconciled based on effects characteristic of a given accident scenario.

Offsite Dose Assessment

Without a sample of the post-accident fluid to be released, there are only three methods for performing offsite dose projections:

- Containment Area Radiation Monitors
- Field Team Data
- Effluent Process Radiation Monitors

Commonly available dose assessment codes used in emergency response, such as MESOREM, are capable of estimating the release source term based on area dose rates in containment. This is typically achieved by calculating the concentration of ambient noble gases required to produce the monitor response using a default mix of radionuclides. Then the concentration of radioiodine is estimated using default ratios of noble gas to iodine. Once the activity concentrations are determined, they are used with the free volume of containment or measured or projected discharge rates and meteorological dispersion models to calculate offsite dose. Variations of this method include the use of hand held survey instruments to measure dose rate on the exterior surface of containment, and possibly including the personnel access hatches.

The major limitation of this method is the extremely large uncertainty created in the determination of the source term that results from the use of default radionuclide distributions and other assumptions about how other nuclides are physically distributed throughout the containment spaces. In addition, uncertainty is also increased by the use of crude (or non-existent) corrections for the effects of shielding inside containment, plateout of radioactive materials on the monitor itself, and the monitor efficiency for detecting various emissions. As a result, the actual activity concentrations may by off by several orders of magnitude.

As part of the emergency response organization, the utilities and local government have the capability to deploy field teams that can measure field dose rates and collect air grab samples, which can be used to determine local iodine concentrations in the field. The limitations of this method for offsite dose assessment and for possible use in core damage assessment are:

- While this method is excellent for determining the actual dose received by people exposed to the release, the data is valid only for a small location. Given the enormous variation in the distribution of radiation possible with atmospheric dispersion, activity concentrations could very significantly over the scale of a few dozen meters, even if the release plume behaved in a straight-line gaussian fashion.

- Similarly, the method can be unreliable. It presupposes that the path of released materials can be determined based on a single wind direction measurement taken at the station. Given the possible variation in wind field away from the plant, the buoyancy of the plume and the associated uncertainty about how the plume will distribute vertically and where it will touch ground, the timing of the release, and the availability of roads, the field team may not be able to collect a representative sample of the effluent. It is not uncommon during emergency response drills and exercises for field teams to miss a simulated, well-behaved plume entirely. This difficulty is compounded if the event occurs at night or during a rain.

- The data is difficult to obtain. In order to collect an iodine sample, the field team must physically enter the radioactive plume, traverse it in the cross-wind direction to determine the centerline location, and then stop and run an air sampling pump for a number minutes. Depending on the magnitude of the release, some field team members may understandably be reluctant to perform this task.

- Field team data has limited value for protective action decision making because by the time the activity can measured in the field, it is too late to implement any protective actions.

- Field team data is not valid for core damage assessment because of the vastly different chemical and physical properties between noble gases and iodines. Given that the two types of radioactive material distribute, chemically react, and deplete in the atmosphere differently, it is highly unlikely that the noble gas to iodine ratio in the field will be remotely similar to what they are in the plant.

- Field data will only be useful for core damage assessment in cases where there is an offsite release of radioactive materials. The data will not be available in events where the containment functions as designed.

In the CE System 80, the normal effluent release pathways from the Plant Vent and the Fuel Building contain process radiation monitors with provisions that allow for sampling of high activity noble gases, iodines, and particulates. This provides the opportunity to determine the distribution of radionuclides in the release; however, there are two obvious drawbacks. First, all station engineering and procedural controls are intended to prohibit the release of radioactive material during an accident. Second, this information has limited value for preemptive protective active decision making because it is too late to implement any protective actions by the time the activity in the effluent can collected and analyzed. Note that non-effluent radiation monitors, such as those on the main steam line used during steam generator tube rupture, are gross activity detectors that provide no information about radionuclide distribution without supplemental grab samples.

Verification of Proper Safety System Operation

A secondary use of PASS data during reactor accidents is to verify that engineered safety features are operating as designed. At considerable expense, the NRC and the nuclear utilities have developed a no less than excellent understanding of power reactor design, the potential failure mechanisms, and the capability of engineered safety feature to reduce offsite radiation exposure. The current regulations ensure that the consequences of the design basis events are minimal and the risks of plant operation to the public are conservative with respect to other industrial applications. Nevertheless, no matter how well thought out the accident scenarios are, there is always the possibility that a heretofore unseen condition or effect might make the accident consequences worse than expected. Therefore, PASS can have considerable value in that it can verify that certain aspects of the accident mitigation are going as designed or alert the operators to conditions that may be outside the design basis.

For example, although not credited in the safety analysis, the retention of radioiodine in the containment sump liquid, instead of the containment atmosphere, will produce significant reduction in actual offsite thyroid dose following a LOCA type event. At PVNGS, the high solubility iodine in the containment sump is ensured by placement of bins of tri-sodium phosphate (TSP) on the floor of containment to raise pH between 7.0 and 8.5. The suitability of this passive engineered safety feature is demonstrated by a calculation that uses reasonable and conservative assumptions. However, there is still a possibility that some unanticipated factor about the design and maintenance of the TSP bins, the solubility of TSP, or the chemical environment of the post-LOCA sump liquid could result in pH outside the desired range. If PASS results could confirm that the pH was within design, this frees the emergency response staff up to address more pressing issues. If pH is not within the design range, then the plant staff needs to take action to restore it.

The System 80 design provides for Hydrogen Recombiners that can maintain the concentration of hydrogen in the containment atmosphere below that required to produce a hydrogen burn or detonation. The decision to place this equipment in service and the evaluation of its performance are based on measurement of hydrogen concentration with the in-line Hydrogen Analyzer. While there is more than reasonable assurance that the Hydrogen Recombiners will perform as designed, there is still a chance that some part of design, fabrication, testing, maintenance, or calibration of the Hydrogen Analyzer may prevent it from correctly measuring or trending hydrogen concentration. The availability of PASS provides an independent method of confirming that the combustible gas control equipment is operating as designed. This either alerts the emergency response staff to an accident mitigation problem or frees them to address other issues.

PASS results can also be used to confirm successful operation of emergency core cooling, containment spray for fission product removal, and reactivity control.

The Case Against PASS

The stated case against PASS is based on six judgments that are unfair simplifications of PASS capability, which falsely devalue its worth in emergency response activities.

First, it is assumed that PASS results are not timely and are therefore not useful for protective action decision making. At PVNGS it is true that collection of all the samples required for core damage assessment and offsite dose projection make take several hours. In rapidly evolving scenarios then, it will be necessary to use other less accurate, but faster methods. In all likelihood, however, the emergency response for a major reactor accident will continue for days, and the relatively long lead time for PASS data will not be a significant impediment to plant assessments and offsite dose calculations.

Second, it is assumed that PASS may not produce accurate results because the associated methods assume that the sample is representative and that the containment is well mixed. While it is true that there are some scenarios, particularly those without containment spray, where post-accident fluids may not be well mixed, however, in those scenarios, the planned alternative to PASS, containment radiation monitors, will not be valid either.

Third, it is assumed that because iodine plates out in the sample lines, PASS results are not capable of providing reliable indication of iodine concentration in containment atmosphere. While iodine plateout is a widely observed and documented effect, it is also true that plateout cannot continue indefinitely. As more and more iodine accumulates in the sample lines, the rate at which iodine is re-suspended into the sample lines increases. If the sample is left in recirculation long enough, the system will come to equilibrium when the rate of plate out is balanced by the rate of re-suspension. At this point, uncertainty associated with the iodine sample measurements due to plateout will be negligible.

Forth, within the context of protective action decision making, it is a widely held -- and mistaken -- belief that recommending evacuation of the public is always a conservative course of action. This philosophy intrinsically devalues PASS by reducing the necessity for accurate core damage assessment and offsite dose projection and calculation. The PASS equipment allows for more accurate assessments and permits independent confirmation of the other methods. If the public will be evacuated based on "any indication" of fuel damage rather than the best assessment of the radiological hazards, these advantages will add little value to the emergency response. However, there can be significant health risks associated with evacuating a local population due to the increased chance of traffic accident injury, especially if the evacuation is done in panic. The risk of a traffic injury or fatality during evacuation is a function of the population density, availability of roads outside the plume pathway, time of day, and prevailing weather conditions. A responsible emergency response organization should only recommend evacuation when the health risk associated with the actual radiation exposure is greater than the risk of evacuation. Therefore, it is desirable to have the best available data, which has been verified to the extent practical, when making protective action decisions.

Fifth, as a consequence of the first and fourth items, many fail to recognize that an emergency response includes more than the initial reaction and first set of protective action recommendations. At some point, as the offsite risk subsides, the utility and state and local governments also need to downgrade the emergency classification and discontinue performance of protective actions. Here, again, a responsible emergency response organization should attempt to minimize the inconvenience and cost to the public by allowing people to return to their homes and business as soon as their health is no longer jeopardized.

Without PASS, dose assessment based on conservative, default nuclide mixes will ensure that this objective will not be met.

Sixth, Reference 5 indicates that PASS is no longer required during emergency response in part because of the development of the Severe Accident Mitigation Guidelines (SAMGs). The SAMGs were intended to provide guidance to the plant operator under severely degraded accident conditions that are outside the plant design and licensing basis. Based on in-plant instrument readings, the core damage state is classified as either "in-vessel" or "ex-vessel," and the status of containment is classified as either intact, bypassed, impaired, or challenged (faulted). Depending on the state of the core and containment, various Candidate High Level Actions (CHLAs) or mitigation strategies are recommended in order of preference. Although the SAMGs have provided great insight into the advantages and disadvantages of various operator actions under degraded conditions, careful review of the CHLAs shows that -- with minor exceptions -- the actions described are already contained in the emergency operating procedures. The proposed action can reduce offsite dose from a severe accident, but the SAMGs do not contain any directions that obviate the need for assessment of the core or offsite dose consequences. Further, the core damage classification scheme is too crude to indicate whether there is significant clad damage, fuel overheat, or pellet melt. This information, which is provided by PASS results, can influence some operator mitigation actions:

- The indication of fuel overheat may mean that the cladding has been embrittled. This might prompt extra care in throttling of core cooling flow rates, the avoidance of re-starting a LPSI pump if not required, or avoidance of simultaneous use of LPSI and CS pumps on shutdown cooling to reduce flow induced vibration.
- High levels of activity in the sump may preclude the use of shutdown cooling as the long-term cooling (LTC) strategy since most of that equipment is located in the Auxiliary Building.
- Indication of significant fuel melt alerts to the operator to possible loss of coolable geometry and the need to monitor individual CETs and maintain vessel water level as high as possible.
- Increasing levels of core damage with all flows within design ranges may indicate that cooling flow is bypassing the core.

Consequently, the SAMGs are not a replacement for PASS in evaluating core cooling. More importantly, the SAMGs do not provide any methods to assess the potential dose consequences that drive the offsite emergency response.

Conclusions

Following an accident at a nuclear power plant, the utility and the state and local governments have the responsibility for making protective action recommendations for the population in the vicinity of the affected station. Because there are risks associated with protective actions, recommendations should be based on due consideration of the relative risk between the action and the radiological health consequences. Such evaluations cannot be performed without reasonable estimations of the potential offsite radiation dose.

While there are methods for core damage assessment and offsite dose assessment that do not require PASS data, these methods have severe limitations with respect to applicability and accuracy. And, without PASS, there is frequently no way to corroborate the results of those analyses. Analyses based on PASS data also have limitations, but, in most accident scenarios, methods utilizing PASS results will provide the most accurate and reliable assessments. In addition, PASS allows for the independent verification that important safety equipment is functioning as designed.

I agree that some of the PASS performance criteria are unnecessarily restrictive and pose burdens that are not commensurate with their benefit, but I contend that deletion of the requirement to obtain radionuclide information or elimination of the system entirely represents a serious degradation of the station's capability to assess and respond to the transient. Having personally observed and participated in emergency response exercises over the last 15 years variously as a core damage analyst, mechanical engineer, health physicist, dose assessment coordinator, and field team member, the value of this information is indisputable. The fact that assessments can be performed in the absence of PASS data does not justify its elimination when the obvious and substantial reductions in the quality of those assessments are considered.

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