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Accident Source Term Methodologies and Corresponding Release Fractions

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General Comment

November 8, 2020

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
Attention: Rulemakings and Adjudications Staff

As the petitioner, I submit the following observations and insights as a public comment to PRM-50-122, "Accident Source Term Methodologies and Corresponding Release Fractions" (NRC-2020-0150).

Further research indicates the errors and omissions of Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," are pervasive and their origins are suspect.

NUREG/CR-5247 (RASCAL) tells us:
Predicting [nuclear accident] doses or consequences . . . requires several steps: (1) predicting the quantity and timing of the release from the plant (source term), (2) predicting the movement

of the plume (transport), and (3) predicting the dose from the plume and predicting the health effects from the dose. Each of these steps requires collection of appropriate data, and data collection and the subsequent computations are subject to uncertainties.

The largest single component of uncertainty is expected in the estimate of the source term. Unanticipated catastrophic containment failure is a case in which the source term could be underestimated by a factor of 1,000,000 if monitor readings are used to estimate the source term. [Emphasis added]

Because the "the accident source term is a fundamental assumption upon which a large portion of the facility [nuclear power plant] design is based," its uncertainty is, arguably, the single most consequential factor that affects the confidence level of nuclear safety. While uncertainties cannot reasonably be eliminated in nuclear accident analyses, it is incumbent upon those performing and overseeing these analyses to objectively evaluate the uncertainties, such that the level of confidence given to these analyses appropriately reflect the level of safety they provide. Herein lies the problem with RG 1.183.

The NRC and the Government Accountability Office (GAO) guidelines require that uncertainties be addressed in regulatory analyses for radiological exposure; however, source term uncertainties are not addressed in RG 1.183 or in the regulatory analyses performed by licensees. Furthermore, because of the conceptual errors identified in RG 1.183, the, already, unknown uncertainties in the resulting radiological exposure analyses are multiplied by an unknown factor. Its conceptually inaccurate and non-conservative assumptions have been integrated into a large portion of nuclear power plant design and a wide range of licensing activities. These false assumptions clearly indicate the safety of nuclear power is uncertainat best.

The intent of this comment is to promote responsible nuclear safety rule-making.

Sincerely,
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Acting expressly as a member of the public.

Attachments

Magnuson Public Comment on PRM-50-122

PRM-50-122

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Research and Observations:

1. MSIVs are NOT PCIVs

As defined by 10 CFR 50.2, MSIVs are part of the Reactor Coolant System (RCS), as such, they are part of the Reactor Coolant pressure Boundary (RCPB). However, main steam isolation valves (MSIVs) on Boiling Water Reactors (BWR) are commonly considered primary containment isolation valves (PCIVs) in plant Technical Specifications and NRC guidance (e.g., NUREG-1433, Rev. 4). Unfortunately, this characterization is misleading and inaccurate because MSIVs are designed to isolate the *reactor vessel* (reactor coolant pressure boundary)—to control the loss of coolant from the reactor vessel and the release of radioactive materials to the environment in an accident.

This mischaracterization did not always exist. Contrary to plant Technical Specifications and NRC guidance, MSIVs are not Primary Containment Isolation Valves (PCIV); they are, in fact, reactor coolant system (RCS)/reactor coolant pressure boundary isolation valves. The designer of most BWR plants (General Electric) called them "*Reactor Vessel Isolation Valves*." The nuclear industry combined MSIVs with PCIVs and called the group, "*Primary Containment Reactor Vessel Isolation Control System (PCRVICS)*" or just "*CRVICS*"—in plant TS and safety analyses (UFSAR). Sometime later, references to MSIVs as "PCRVICS" or "CRVICS" were systematically removed from plant TS, UFSARs and NRC guidance. Thereafter, MSIVs have been mischaracterized as PCIVs. Nevertheless, stray

references to PCRVICS and CRVICS still exist (e.g., Limerick LER 01-001-00, Fitzpatrick LER 87-021, Fermi UFSAR Revision 21).

Because the conceptually inaccurate description of BWR MSIVs has been deeply imbedded in NRC regulations and plant licensing bases, there is a broad range of generic safety issues.

2. MSIV Leakage Technical Specifications are Inadequate

As an apparent consequence of this mischaracterization, BWR plant technical specification surveillance requirements for MSIV leakage are inadequate because MSIVs are incorrectly tested at a pressure corresponding to the peak primary containment accident pressure—instead of the corresponding peak *reactor pressure*. Therefore, measured MSIV leakage rates at BWR plants are non-conservative and grossly inaccurate. This negates the intent of the technical specification limit of MSIV leakage which is presumed to be conservatively set to ensure that offsite dose consequences of design basis accidents are a small fraction of the regulatory limits in 10 CFR Part 100.

Because of the inaccurate MSIV TS SR test pressure, in the event of an accident, the radiological release through (closed) MSIVs will be much greater than that assumed in design basis dose calculations (AST or other). Given the current margins in these calculations, radiation doses to the public and control room operators will likely exceed federal limits.

Because MSIV leakage results in a loss of reactor coolant, inaccurate and nonconservative accident dose analyses (e.g., LOCA, MSLB), the current MSIV TS SR are inimical to the health and safety of the public.

NRC safety evaluations of License Amendment Requests that accept or acknowledge that “the MSIVs are functionally part of the primary containment boundary” are conceptually inaccurate and fail to satisfy 10 CFR 50.92(c) because they create significant hazards. License Amendment Requests approved under this false assumption (1) involve a significant increase in the consequences of an accident previously evaluated; (2) create the possibility different kind of accident from any accident previously evaluated; and (3) involve a significant reduction in a margin of safety.

3. Appendix J to Part 50 is also Conceptually in Error and Misapplied to MSIVs and other Reactor Vessel Isolation Valves

Appendix J to Part 50—Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors:

> *H. "Type C Tests" means tests intended to measure containment isolation valve leakage rates. The containment isolation valves included are those that:*
>
> *1. Provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation, such as purge and ventilation, vacuum relief, and instrument valves;*

2. Are required to close automatically upon receipt of a containment isolation signal in response to controls intended to effect containment isolation;

3. Are required to operate intermittently under post-accident conditions; and

4. Are in main steam and feedwater piping and other systems which penetrate containment of direct-cycle boiling water power reactors.

As previously described, MSIVs are not containment isolation valves. Main steam lines do not provide a direct connection between the inside and outside atmospheres of the primary reactor containment under normal operation. Furthermore, the reactor coolant pressure boundary extends to outermost isolation valves in feedwater, HPCI, RCIC, Reactor Water Cleanup and other reactor coolant system piping. Similar to RG 1.183, Appendix J is conceptually in error and misapplied to these reactor vessel isolation valves (reactor pressure boundary isolation valves).

4. MSIV LLRT Failures are Failures of the Reactor Coolant (System) Pressure Boundary

MSIV Local Leak Rate Test failures are failures of the Reactor Coolant System Pressure Boundary; however, these common failures are not monitored by the Reactor Oversight Process (ROP) as "*Reactor Coolant System Leakage"* or identified in NRC Inspection Manual Chapter 0308, Attachment 1.

MSIV Local Leak Rate Testing failures are violations of plants technical specifications and are often reported under 10 CFR 50.73 (a)(2)(ii) "because an event occurred which resulted in the degradation of one of the plant's principal safety barriers." Regrettably, these License Event Reports do not recognize or acknowledge that these failures are failures of the reactor coolant pressure boundary. As such, the safety consequence of these failure is routinely minimized. These failures of the RCS/RCPB have not been considered "significant conditions adverse to quality"; therefore, measures to assure that corrective actions are taken to preclude repetition have NOT been implemented by the licensees or enforced by the NRC as required by Appendix B to Part 50, XVI. Corrective Action.

Furthermore, contrary to 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," reported MSIV LLRT failures have not been appropriately categorized as maintenance-preventable functional failures (MPFFs), as such, maintenance that would prevent failures of the reactor coolant pressure boundary is not performed.

Maintenance preventable failures of the reactor coolant pressure boundary are also contrary to Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10, Part 50, "Domestic Licensing of Production and Utilization Facilities," of the Code of Federal Regulations (10 CFR Part 50). General Design Criterion (GDC) 14, "Reactor Coolant Pressure Boundary," requires that licensees or applicants design, fabricate, erect, and test the reactor coolant pressure boundary (RCPB) so as to ensure an extremely low probability of abnormal leakage.

5. There is Only One Physical Fission Product Barrier

Contrary to NRC and industry publications, there are not necessarily three[1] physical barriers between reactor core fission products (millions of Curies) and the environment.

MSIVs are part of the Reactor Coolant System and pressure-containing components of the Reactor Coolant Pressure Boundary; however, in an accident, closed MSIVs will not prevent the release of fission products to the environment. Therefore, the Reactor Coolant System is not an effective physical barrier. AST LOCA dose consequence analyses indicate MSIV leakage is a significant contributor to operator/main control room doses. Therefore, the failure of only one physical barrier—approximately 0.029 inches of fuel cladding—will result in a significant radiological release to the environment.

Similarly, in any spent fuel pool accident, the fuel cladding provides the only physical barrier between fission products and the environment, and in a design basis Fuel Handling Accident that one barrier is lost. As we have learned from NRC Information Notice No. 90-08*: "KR-85 Hazards From Decayed Fuel*," no amount of water shielding in spent fuel pools will protect workers in the area or prevent the release of nobel gasses to the environment in the event that spent fuel cladding is ruptured from of mechanical damage or overheating.

[1] Because of fuel cladding gap releases and the effects of high burnups which can reduce fuel pellets to powder, the integrity of the fuel pellet form is no longer considered a physical fission product barrier.

6. "Recently" Irradiated Fuel

It should be noted that license amendments that adopted the alternative source term methodology, as prescribed in Title 10 to the Code of Federal Regulations Section 50.67, also revised the technical specification sections associated with the implementation of Technical Specification Task Force (TSTF) - 51 traveler, which provided relaxation of certain requirements during movement of irradiated fuel.

> *"The purpose of the TSTF-51 TS changes is to establish a point where OPERABILITY of ESFs typically used to mitigate the consequences of a FHA are no longer required to meet the SRP guidance on offsite dose limits (i.e., less than 25 percent of the 10 CFR Part 100, Reactor Site Criteria," limits or the limits specified in 10 CFR 50.67). Specifically, the proposal identifies that only "recently" irradiated fuel contains sufficient fission products to require OPERABILITY of the accident mitigation features to meet the accident analysis assumptions. Therefore, the APPLICABILITY requirements for the associated mitigation features (including the electrical support systems) are revised. The requested changes would eliminate TS requirements for ESFs during core alterations by deleting "During CORE ALTERATIONS" from APPLICABILITY. The requested change also adds "recently" to "irradiated fuel" to revise APPLICABILITY to "During movement of recently irradiated fuel." The affected TS Limiting Conditions for Operation (LCO) required ACTION statement to immediately suspend movement of irradiated fuel assemblies in secondary containment, when the LCO is not met, is also revised to require such action only when recently irradiated fuel assemblies are moved."*

TSTF-51 (Revision 2), which was approved by the NRC on October 13,1999. It predates RG 1.183 (July 2000) and conflicts with NRC, "*Information Notice No. 90-08: KR-85 Hazards From Decayed Fuel*" (February 1, 1990) which states:

> *"Analysis of hypothetical accidents involving decayed spent fuel has focused attention on potential difficulties that could be associated with the exposure of onsite personnel to an accidental release of Kr-85. Kr-85 is a noble gas fission product that is present in the gaps between the fuel pellets and the cladding. It has a 10.76-year half-life, and, as a result of the considerably shorter half-lives of virtually all other gaseous fission products (I-129 being the exception, but in low abundance), Kr-85 becomes increasingly the dominant nuclide in the accident source term for gap releases as decay times increase. After 2 weeks of decay, Kr-85 is a significant nuclide in the source term, and after 190 days of decay, it is the predominant gaseous nuclide for a gap release. The unusual decay characteristics of Kr-85 give cause for focusing attention on the onsite consequences of a gap release from decayed fuel."*
>
> *"Kr-85 emits beta radiation with a maximum energy of 0.67 MeV for 99.6 percent of the decays and 0.51 MeV gamma radiation for 0.4 percent of the decays. Consequently, direct exposure to this gas would result in a dose to the skin approximately 100 times the whole-body dose. Analysis of the relative consequences (in terms of radiological doses) of a cask-drop accident as a function of decay time of the fuel is illustrated in Figure 1. In the event of a serious accident involving decayed spent fuel, protective actions would be needed for personnel on site, while offsite doses (assuming an exclusion area radius of 1 mile from the plant site) would be well below the Environmental Protection Agency's Protective Action Guides. Accordingly, it is important to be able to properly survey and monitor for Kr-85, and to assess the skin dose to workers who could be exposed to Kr-85 in the event of an accident with decayed spent fuel."*
>
> *"Licensees may wish to reevaluate whether Emergency Action Levels specified in the emergency plan and procedures governing decayed fuel-handling activities appropriately focus on concern for onsite workers and Kr-85 releases in areas where decayed spent fuel accidents could occur, for example, the spent fuel pool working floor. Furthermore, licensees may wish to determine if emergency plans and corresponding implementing procedures address the means for limiting radiological exposures of onsite personnel who are in other areas of the plant. Among other things, moving onsite personnel away from the plume and shutting off building air intakes downwind from the source may be appropriate."*

The following reference provides additional insights and indicates that TSTF-51 inappropriately allowed licensees to eliminate the technical specification requirements for spent fuel pool area gamma radiation monitors because they would not detect the beta radiation emitted by Kr-85 in a Fuel Handling Accident.

"Comments on Direct Final Rule Change to 10 CFR 50.68" (Serial: RNP-RA/06-0121 USNRC) states:

> *The typical design of a nuclear power plant includes one or more gamma sensitive Area Radiation Monitors (ARMs) located in the area above the SFP. While loading a cask in the SFP, there will be approximately 23 feet of water between the ARM and a potential criticality event in the cask. With this significant amount of intervening shielding, these ARMs will not respond to the direct radiation resulting from a criticality event. The criticality event could result in cladding*

damage and the release of the fuel gap fission products. However, fuel being loaded into dry storage casks will have decayed for at least 3 years, therefore, the only fission product released from the fuel rod gap to the area above the SFP that is of any dose significance would be Kr-85. Kr-85 is essentially a beta emitter (only 1 gamma every 250 disintegrations) and hence the ARMs, which are only sensitive to gamma radiation, would likely not alarm. However, the airborne concentrations of Kr-85 could represent a skin dose hazard to the personnel by the SFP (see NRC Information Notice 90-08). These ARMs cannot meet the sensitivity requirements for criticality monitors as specified in 10 CFR 70.24(a)(1). 10 CFR 72 does not provide similar specific requirements for a criticality monitoring system. If the requirements for criticality monitoring to meet 10 CFR 72.124(c) are more general (e.g., a system that would warn of a radiation hazard to personnel), then the current ARMs would not meet that requirement either due to the Kr-85 impact.

The SFP ARMs cannot be considered criticality monitors because they will not respond to a criticality event. This was the reason nuclear power plants had to apply for exemptions to 10 CFR 70.24 and the reason 10 CFR 50.68 was written. The wording in 10 CFR 50.68 implies that these ARMs are not criticality monitors, as the rule states that in lieu of maintaining a criticality monitoring system, the licensee must meet a number of criteria, one of which is to maintain a radiation monitoring system in the fuel handling area. Licensees have taken credit for the SFP ARMs to meet this requirement. If these ARMs could be considered criticality monitors then 10 CFR 50.68 would not be required. If the interpretation of the requirements of 10 CFR 72.124(c) for underwater monitoring, as provided in the Technical Evaluation, are not corrected, then licensees may have to file exemption requests to 10 CFR 72.124(c).

IN 90-08 appears to have identified a generic safety issue in 1990 that still exists today.

7. MSIVs are Leak Tight *only* in an AST MSLB

RG 1.183, LOCA *ASSUMPTIONS ON MAIN STEAM ISOLATION VALVE LEAKAGE IN BWRs* states:

For BWRs, the main steam isolation valves (MSIVs) have design leakage that may result in a radioactivity release. The radiological consequences from postulated MSIV leakage should be analyzed and combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. The following assumptions are acceptable for evaluating the consequences of MSIV leakage.

All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.

MISVs were not designed to leak as RG 1.183 would have you believe. As previously explained, they leak because of (normal wear from high pressure steam flow and ineffective preventative maintenance practices. Reference NRC Staff Evaluation of the NUREG—1285 General Electric Company Nuclear Reactor Study ("Reed Report"):

PRM-50-122

> *"4.3 Main Steam Isolation Valve Leak Tightness Issue: The issue of leak tightness of main steam isolation valves (MSIVs) was identified in the Reed Report in the section on Mechanical Systems and Equipment, but was not discussed in the GE status report provided in 1978. Main steam isolation valves (MSIVs) have been notorious for leaking at high rates when they are tested during the 18-month leak tightness testing that is generally required by the technical specifications. Most plants have a technical specification leak rate limit of 11.5 standard cubic feet per hour (scfh) per valve. At some plants the as-found leak rate has been as high as 4500 scfh.*
>
> *MSIV leak tightness was a concern in 1975, and it is still a concern that has not been fully resolved. The BWR Owners Group (BWROG) formed a committee to evaluate this same issue independently, with GE giving technical support to the BWROG committee. THIS COMMITTEE GENERALLY FOUND THAT THE HIGH LEAKAGE RATES WERE ATTRIBUTABLE TO VALVE MAINTENANCE PRACTICES. For those plants that have adopted the BWROG recommendations resulting from their evaluation, the as-found MSIV leak rates have generally been within the plant-specific technical specification limit . . . For example, Peach Bottom 3, had typical as-found leak rates of over 3000 scfh for each of the MSIVs. After following the BWROG recommendations, the next as-found leak rates were found to be less than 11.5 scfh for seven of the eight MSIVs and approximately 14.7 scfh for the eighth MSIV. THIS DEMONSTRATES THAT THE MSIVS CAN BE MAINTAINED WITHIN THEIR RESPECTIVE TECHNICAL SPECIFICATION LEAKAGE LIMITS . . ."* [Emphasis added.]

RG 1.183 reasonably assumes MSIVs will leak for 30 days in a LOCA; however, it tergiversates and illogically assumes the radiological release from a Main Steam Line Break (MSLB) will be terminated when the MSIVs close (approximately 10 seconds). RG 1.183 allows licensees to assume "*The total mass of coolant released should be assumed to be that amount in the steam line and connecting lines at the time of the break plus the amount that passes through the valves prior to closure."*

This false assumption results in the under calculation of pre-accident doses that are used to determine the level of protection necessary to protect control room operators and the public from overexposures in the event of a MSLB accident.

Here it is important to recognize that MSIV leakage is the single most significant contributor to operator/main control room doses and the most limiting regulated dose in accident (MSLB) analyses, and consider that Control Room Habitability regulatory guidance that required physical plant modifications to protect control room operators, lagged the identification of MSIV leakage concerns by over twenty years but coincided with the issuance of RG 1.183 / 10 CFR 50.67 that introduced the TEDE dose limits.

Furthermore, contrary to RG 1.196, "*Control Room Habitability at Light-Water Nuclear Power Reactors*, MSLB dose analyses are not reperformed in all licensing activities associated with MSIV leakage requirements. RG 1.196 states:

> *"In determining the limiting condition for potential radiological accidents, it should not be presumed that the LOCA is the limiting accident because it has the largest initial source of activity. Other accidents, e.g., fuel handling accidents, may produce larger control room operator doses because the manner in which the CRHSs respond may provide less protection to the operators."*

8. Post-Accident Dose Analyses Will Not Prevent Accidents or Overexposures

As its name suggests, RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors is intended for pre-accident dose analyses; however, as it states, can be applied to post-accident dose assessments as required by 10 CFR 50.47 Emergency Plans and Appendix E to Part 50, but is not. Instead, the nuclear industry continues to use the RASCAL source terms and methodologies, that unlike RG 1.183, have been periodically updated to reflect insights such as those learned from the accident at Fukishima (NUREG- 1940 RASCAL 4).

The RASCAL source terms are more conservative and do not contain the conceptual errors that SAND2008-6601 identified in RG 1.183. This inconsistent application of accident source terms does not appear to be in the best interest of the health and safety of the public.

REFERENCES

10 CFR 50.2: *Reactor coolant pressure boundary means all those pressure-containing components of boiling and pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps, and valves, which are:*
(1) Part of the reactor coolant system
For nuclear power reactors of the direct cycle boiling water type, the reactor coolant system extends to and includes the outermost containment [sic] isolation valve in the main steam [MSIVs] *and feedwater piping.*

APPENDIX A TO PART 50—GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

> *GDC 14—Reactor coolant pressure boundary [e.g., MSIVs]. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.*
>
> *GDC 15—Reactor coolant system [e.g., MSIVs] design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to*

assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

GDC 30—Quality of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary [e.g., MSIVs] shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

GDC 31—Fracture prevention of reactor coolant pressure boundary. The reactor coolant pressure boundary [e.g., MSIVs] shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

GDC 32—Inspection of reactor coolant pressure boundary. Components which are part of the reactor coolant pressure boundary [e.g., MSIVs] shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

GDC 55: Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment [e.g., MSIVs] shall be provided with containment isolation valves . . .

GDC 56: Primary containment isolation. Each line that connects directly to the containment atmosphere [not MSIVs] and penetrates primary reactor containment shall be provided with containment isolation valves . . .

APPENDIX B TO PART 50—QUALITY ASSURANCE CRITERIA FOR NUCLEAR POWER PLANTS AND FUEL REPROCESSING PLANTS

III. Design Control

Measures shall be established to assure that applicable regulatory requirements and the design basis, as defined in § 50.2 and as specified in the license application, for those structures, systems, and components to which this appendix applies are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled. Measures shall also be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the structures, systems and components.

Measures shall be established for the identification and control of design interfaces and for coordination among participating design organizations. These measures shall include the establishment of procedures among participating design organizations for the review, approval, release, distribution, and revision of documents involving design interfaces.

The design control measures shall provide for verifying or checking the adequacy of design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program. The verifying or checking process shall be performed by individuals or groups other than those who performed the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, it shall include suitable qualifications testing of a prototype unit under the most adverse design conditions. Design control measures shall be applied to items such as the following: reactor physics, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for inservice inspection, maintenance, and repair; and delineation of acceptance criteria for inspections and tests.

V. Instructions, Procedures, and Drawings

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

VI. Document Control

Measures shall be established to control the issuance of documents, such as instructions, procedures, and drawings, including changes thereto, which prescribe all activities affecting quality. These measures shall assure that documents, including changes, are reviewed for adequacy and approved for release by authorized personnel and are distributed to and used at the location where the prescribed activity is performed. Changes to documents shall be reviewed and approved by the same organizations that performed the original review and approval unless the applicant designates another responsible organization.

XI. Test Control

A test program shall be established to assure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures which incorporate the requirements and acceptance limits contained in applicable design documents. The test program shall include, as appropriate, proof tests prior to installation, preoperational tests, and operational tests during nuclear power plant or fuel reprocessing plant operation, of structures, systems, and components. Test procedures shall include provisions for assuring that all prerequisites for the given test have been met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. Test results shall be documented and evaluated to assure that test requirements have been satisfied.

XVI. Corrective Action

PRM-50-122

Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

(1980) NUREG-0737 CLARIFICATION OF TMI ACTION PLAN REQUIREMENTS

The design-basis-accident (DBA) radiation source term should be for the loss-of-coolant accident LOCA containment leakage and engineered safety feature (ESF) leakage contribution outside containment as described in Appendix A and B of Standard Review Plan Chapter 15.6.5. In addition, boiling-water reactor (BWR) facility evaluations should add any leakage from the main steam isolation valves (MSIV) (i. e., valve-stem leakage, valve seat leakage, main steam isolation valve leakage control system release) to the containment leakage and ESF leakage following a LOCA. This should not be construed as altering the staff recommendations in Section D of Regulatory Guide 1.96 (Rev. 2) regarding MSIV leakage-control systems.

Other DBAs should be reviewed to determine whether they might constitute a more-severe control-room hazard than the LOCA. In addition to the accident-analysis results, which should either identify the possible need for control-room modifications or provide assurance that the habitability systems will operate under all postulated conditions to permit the control-room operators to remain in the control room to take appropriate actions required by General Design Criterion 19, the licensee should submit sufficient information needed for an independent evaluation of the adequacy of the habitability systems. Attachment 1 lists the information that should be provided along with the licensee's evaluation.

(1982) Reference INFORMATION NO. 82-23: MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE

IE [NRC Office of Inspection and Enforcement] has completed a survey of MSIV performance at BWRs for the years 1979 through 1981. IE found that 19 of 25 operating BWRs had MSIVs which failed to meet, during one or more surveillance tests, the limiting condition for operation (LCO) which specifies the maximum permissible leak rate. The number of MSIV test failures exceeded 151 and occurred with MSIVs supplied by all three MSIV vendors, i.e., Atwood & Morrill, Crane, and Rockwell.

Measured leak rates which exceeded the LCO ranged from greater than 11.5 standard cubic feet per hour (scfh) to 3427 scfh. Twelve stations had 57 MSIV tests with results greater than 11.5 scfh and less than 100 scfh, and five stations (nine units) had 66 MSIV tests with results between 100 and 3500 scfh. Four other licensees had more than 24 test failures but did not measure, estimate, or report the magnitudes of the leak rates.

This information indicates that some MSIVs may not adequately limit release of radioactivity to the environment if called upon to do so. NRC is considering the need for improved MSIV maintenance, more frequent MSIV testing or installation of leakage control systems.

(1986) Availability of NUREG-1169, "Technical Findings Related to Generic Issue C-8; Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods" (Generic Letter No. 86-17)

(1990) IN 90-08, KR-85 HAZARDS FROM DECAYED FUEL

(1989, 1992) NUREG/CR-5247, RASCAL Version 2.1 User's Guide

> *The Radiological Assessment System for Consequence Analysis (RASCAL) (Athey et al. 1989, 1992) is a set of personal computer-based tools. RASCAL Version 2.1 contains tools to estimate source term, atmospheric transport, and dose from a radiological accident (ST-DOSE), to estimate dose from field measurements of radionuclide concentrations (FM-DOSE), and to compute decay of radionuclides (DECAY). RASCAL was developed for use by U.S. Nuclear Regulatory Commission (NRC) personnel who report to the site of a nuclear accident to conduct an independent assessment of dose projections.*

> *Predicting* [nuclear accident] *doses or consequences . . . requires several steps: (1) predicting the quantity and timing of the release from the plant (source term), (2) predicting the movement of the plume (transport), and (3) predicting the dose from the plume and predicting the health effects from the dose. Each of these steps requires collection of appropriate data, and data collection and the subsequent computations are subject to uncertainties.*
>
> *The largest single component of uncertainty is expected in the estimate of the source term. Unanticipated catastrophic containment failure is a case in which the source term could be underestimated by a factor of 1,000,000 if monitor readings are used to estimate the source term. For lesser (non-core damage) accidents in which the total release is through a monitored pathway and consists mostly of noble gases, the source term uncertainty can be reduced. However, the transport and dose uncertainties would remain unchanged.* [Emphasis added]

(1994) SECY-94-302, SOURCE TERM-RELATED TECHNICAL AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND PASSIVE LIGHT-WATER-REACTOR DESIGNS

(1998) 10 CFR 50.68 CRITICALITY ACCIDENT REQUIREMENTS

> (*6) Radiation monitors are provided in storage and associated handling areas when fuel is present to detect excessive radiation levels and to initiate appropriate safety actions.* [63 FR 63130, Nov. 12, 1998; as amended at 71 FR 66648, Nov. 16, 2006]

PRM-50-122

(2000) REGULATORY GUIDE 1.183, “ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS”

(2001) NUCLEAR ENERGY INSTITUTE, “CONTROL ROOM HABITABILITY ASSESSMENT GUIDANCE,” NEI 99-03, REVISION 0

(2003) REGULATORY GUIDE 1.194, “ATMOSPHERIC RELATIVE CONCENTRATIONS FOR CONTROL ROOM RADIOLOGICAL HABITABILITY ASSESSMENTS AT NUCLEAR POWER PLANTS”

(2003) REGULATORY GUIDE 1.195 USNRC, “METHODS AND ASSUMPTIONS FOR EVALUATING RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS AT LIGHT-WATER NUCLEAR POWER REACTORS”

(2003, Revision 0) REGULATORY GUIDE 1.196 CONTROL ROOM HABITABILITY AT LIGHT-WATER NUCLEAR POWER REACTORS

> *The primary design function of CRHSs is to provide a safe environment in which the operator can keep the nuclear reactor and auxiliary systems under control during normal operations and can safely shut down these systems during abnormal situations to protect the health and safety of the public and plant workers. If the control room is not habitable or the response of the operator is impaired during an accident, there could be increased consequences to public health and safety. It is important for the operators to be confident of their safety in the control room to minimize errors of omission and commission. The Regulatory Positions below provide methods acceptable to the NRC staff for ensuring that the public and the control room operators are protected.*
>
> *Over the facility’s lifetime the licensing bases change. The staff may have reviewed and approved the licensing bases of facilities licensed before the issuance of this guide. The original licensing bases may have been submitted as part of the construction permit application. Licensees may have modified them in response to NRC questions. In addition, the licensing bases were part of the application for the OL (FSAR). Depending on the plant vintage, licensees may have modified their licensing bases in response to TMI Action Item III.D.3.4. Amendments to the OL may have resulted in changes to the licensing bases of the CRHSs. Licensees should review the applicable plant changes to their licensing bases to determine the current bases.*
>
> *A group of reactors received their construction permits or OLs before the GDCs were promulgated. During this time, proposed GDCs (sometimes called “Principal Design Criteria”) were published in the Federal Register for comment. These proposed GDCs addressed CRH. Although facilities may have been licensed before the promulgation of the GDCs, licensees may have committed to the form of the GDCs as they existed at the time of licensing. A review of the record associated with the construction permit and OL proceedings should confirm whether licensees made such a commitment. Therefore, licensees that received their construction permits or OLs before the GDCs were promulgated should review their commitments to the draft form of the GDC to understand their CRH licensing bases.*

Over the facility's lifetime the licensing bases change. The staff may have reviewed and approved the licensing bases of facilities licensed before the issuance of this guide. The original licensing bases may have been submitted as part of the construction permit application. Licensees may have modified them in response to NRC questions. In addition, the licensing bases were part of the application for the OL (FSAR). Depending on the plant vintage, licensees may have modified their licensing bases in response to TMI Action Item III.D.3.4. Amendments to the OL may have resulted in changes to the licensing bases of the CRHSs. Licensees should review the applicable plant changes to their licensing bases to determine the current bases. A group of reactors received their construction permits or OLs before the GDCs were promulgated. During this time, proposed GDCs (sometimes called "Principal Design Criteria") were published in the Federal Register for comment. These proposed GDCs addressed CRH. Although facilities may have been licensed before the promulgation of the GDCs, licensees may have committed to the form of the GDCs as they existed at the time of licensing. A review of the record associated with the construction permit and OL proceedings should confirm whether licensees made such a commitment. Therefore, licensees that received their construction permits or OLs before the GDCs were promulgated should review their commitments to the draft form of the GDC to understand their CRH licensing bases.

Consistent with Regulatory Position 2.2.1, licensees should ensure that their assumed control room inleakage input value used in any accident calculations or evaluations (Regulatory Positions 2.4 and 2.5) are validated by the test methods provided in Regulatory Guide 1.197.

Unless a facility relies on a common control room isolation process for all types of radiological accidents, the limiting accident may not be obvious. There are several reasons for this:

• The inleakage characteristics of the envelope may vary with the CRHS's response to an accident.

• The mitigative equipment used to reduce the radioactivity released to the environment may vary with the accident.

• The location of the release points for the various accidents relative to the control room intakes may result in less favorable atmospheric dispersion and higher magnitude intake concentrations.

Licensees should factor all the potential differences in accidents and the CRHS's performance in order to determine the limiting condition.

Licensees should calculate control room operator doses for the methodology and accidents identified in Regulatory Guide 1.195 (Ref. 5) or Regulatory Guide 1.183 (Ref. 6). For CREs under construction, the control room operators' doses should be based on expected CRHS performance values. When the envelope and associated ventilation systems are operational, the inleakage value should be determined using Regulatory Guide 1.197 (Ref. 4).

Some licensees were allowed to leave TMI Action Item III.D.3.4 actions open until the alternative source term rulemaking and regulatory guidance were published. These actions were completed with the issuance of 10 CFR 50.67 and Regulatory Guide 1.183 (Ref. 6). The

PRM-50-122

Regulatory Positions in this regulatory guide on control room habitability provide methods acceptable to the NRC staff for closing open TMI Action Item III.D.3.4 actions.

(2003) REGULATORY GUIDE 1.197 “DEMONSTRATING CONTROL ROOM ENVELOPE INTEGRITY AT NUCLEAR POWER REACTORS”

(2007, Revision 1) REGULATORY GUIDE 1.196 CONTROL ROOM HABITABILITY AT LIGHT-WATER NUCLEAR POWER REACTORS

Control Room Acceptance Criteria: The following guidelines may be used in lieu of those provided in SRP 6.4 (Ref. 14) when showing compliance with the dose guidelines in GDC-19 of Appendix A to 10 CFR Part 50. The following guidelines relax the thyroid and skin acceptance criteria from that given in SRP 6.4.

Whole body 5 rem

Thyroid 50 rem

Skin 50 rem[12]

[12]Credit for the beta radiation shielding afforded by special protective clothing and eye protection is allowed if the applicant commits to their use during severe radiation releases. However, even though protective clothing is used, the calculated unprotected skin dose is not to exceed 75 rem. These limits are design criteria and are not to be interpreted as acceptable occupational doses.

(2008) SAND2008-6601 ANALYSIS OF MAIN STEAM ISOLATION VALVE LEAKAGE IN DESIGN BASIS ACCIDENTS USING MELCOR 1.8.6 AND RADTRAD

(2008) SECY-08-172, DENIAL OF PETITION FOR RULEMAKING PRM-50-87 CONCERNING CONTROL ROOM HABITABILITY RADIOLOGICAL DOSE REQUIREMENTS AS GOVERNED BY REGULATIONS SPECIFIED IN APPENDIX A TO 10 CFR PART 50 AND IN 10 CFR 50.67

On May 17, 2007, Mr. Crandall submitted a PRM (PRM-50-87) requesting that the U.S. Nuclear Regulatory Commission (NRC) amend Appendix A, “General Design Criteria for Nuclear Power Plants” to Title 10, Part 50, “Domestic Licensing of Production and Utilization Facilities,” of the Code of Federal Regulations (10 CFR Part 50) and 10 CFR 50.67, “Accident source term.” Specifically, the petitioner requested to delete the 5 rem whole body dose limit specified in General Design Criterion (GDC) 19, “Control Room,” of Appendix A to 10 CFR Part 50 and the 0.05 sievert (Sv) (5 rem) total effective dose equivalent (TEDE) limit specified in both GDC 19 and 10 CFR 50.67 (b)(2)(iii). The petitioner stated that the current deterministic radiological dose requirements for control room habitability have resulted in several negative safety consequences including an increased risk to public safety.

The NRC regards the radiological dose standards, 5 rem TEDE in 10 CFR 50.67 and 5 rem whole body in GDC 19, as performance-based regulations. Performance-based regulations do not provide prescriptive requirements and, therefore, do not require licensees to use specific designs or methodologies to comply with the regulations. However, the NRC does provide regulatory guidance to licensees that includes acceptable designs and methodologies for

demonstrating compliance with the regulations. The use of the guidance is optional, and licensees are free to propose alternative means of complying with the NRC's regulations.

The performance-based control room dose criterion is designed such that an acceptable level of control room habitability will be maintained even under the maximum credible accident scenario. The NRC has determined that providing an acceptable level of control room habitability for design-basis events is necessary to provide reasonable assurance that the control room will continue to be effectively manned and operated to mitigate the effects of the accident and protect public health and safety. By removing the acceptance criteria of 5 rem, a regulatory basis will no longer exist, and would not support the Commission's policy regarding performance-based regulations.

Based upon its review of the petition and the comments submitted, the NRC staff has determined that the conclusions upon which the petitioner relies do not substantiate a basis to eliminate the control room radiological dose acceptance criteria from current regulations as requested. Accordingly, the staff recommends denying the PRM and requests Commission approval to do so and publish the Federal Register notice (Enclosure 1) of the denial.

1. *The petitioner stated that because the primary objective of control room habitability is to ensure continuous occupancy, the primary focus should be on minimizing whole body doses from noble gases. He stated that some common control room designs, such as the filtered air intake pressurization design, focus on compliance with existing dose criteria. He concluded that the current requirements and operational criteria focus on minimizing the thyroid dose at the expense of increasing the whole body dose from noble gases which increases the probability that the control room will require evacuation.*

The NRC reviewed the petitioner's concern regarding the increase in whole body dose from noble gases, which he believes results from the intentional intake of filtered air into the control room under design-basis accident (DBA) conditions. The NRC agrees that a relatively small increase in whole body dose due to noble gases may result from the intake of filtered air into the control room. However, this small increase in dose would not increase the probability of a control room evacuation. Therefore, operators would be able to monitor plant indications and take appropriate accident mitigating actions from the control room, and there would be no increase in risk to public health and safety. The NRC's conclusion is based on a review of several existing DBA control room dose analyses that determined the impact on whole body dose resulting from filtered air intake pressurization to the control room. The NRC performed parametric evaluations and determined that while filtered air intake pressurization may result in a small addition to the control room whole body dose from noble gases, the increase is more than

offset by the reduction in thyroid dose and TEDE from inhalation of radioactive particulates, such as iodine.

Based upon its analyses, the NRC does not agree with the petitioner's assertion regarding the negative safety impact of providing filtered intake flow into the control room. The NRC's performance-based criterion in GDC 19 requires that an applicant provide a control room habitability design that meets the specified dose criterion. Although NRC regulatory guidance provides examples of acceptable design approaches, the approach used to meet the criterion is largely under the control of an applicant. In order to meet this requirement, many licensees have chosen to incorporate filtered air intake pressurization into their control room emergency ventilation designs to reduce the cumulative dose to operators during a DBA. The purpose of providing filtered air intake pressurization flow is to establish positive pressure in the control room relative to the adjacent areas, thereby reducing the quantity of unfiltered air inleakage. Limiting unfiltered inleakage significantly reduces the thyroid dose from inhalation.

The petitioner based his assertion on the assumption that filterable activity is not likely to be a significant contributor to dose in a reactor accident. As an example, the petitioner used the March 1979 Three Mile Island Unit 2 accident. Since the accident, the NRC has expended considerable resources to better define the expected quantity and distribution of activity that could be released during a major reactor accident. As a result of this research, the NRC promulgated 10 CFR 50.67 on December 23, 1999 (64 FR 72001). Under 10 CFR 50.67, a licensee can apply for a license amendment to adopt an alternative source term (AST) that reflects a more realistic assessment of the timing of the release and the quantity and distribution of activity that could be released during a major accident hypothesized for purposes of design analyses. Many licensees have used this approach to comply with NRC regulations governing control room dose.

In addition, 10 CFR 50.67 revised the control room dose criterion from a 5 rem whole body dose, or its equivalent to any organ, to a 5 rem TEDE. The relatively low thyroid organ weighting factor, as defined in 10 CFR 20.1003, "Definitions," and used in the calculation of TEDE, allows for a significant reduction in the controlling aspects of the thyroid dose, which normally governed compliance with control room dose guidelines.

The NRC has significantly improved the accuracy of the source term and dose methodology used in design-basis dose consequence analyses. The updated source term and dose methodology address the petitioner's concerns regarding the emphasis on thyroid dose in control room habitability analyses.

> *The petitioner noted that the dose from increased iodine concentration can be mitigated by use of potassium iodide (KI) or respiratory protection, but the current regulations do not permit these mitigation measures to be used in design analyses.*

The NRC agrees that KI or Self-Contained Breathing Apparatuses (SCBAs) do have merit as short-term compensatory measures. However, the potential medical complications of KI and the potential adverse impacts to human performance of SCBAs make these measures unsuitable for long-term use. Further, the NRC's policy of ensuring that process or other engineering controls are in place instead of relying on the use of personal protective equipment is clearly set forth in 10 CFR 20.1701, "Use of process or other engineering controls" and 10 CFR 20.1702,

"Use of other controls." This policy is consistent with the recommendations of international and national radiation protection committees as described in Paragraph 167 of the International Commission on Radiological Protection (ICRP) Publication 26.

Paragraph 167 of ICRP Publication 26 recommends that "[a]s far as is reasonably practicable, the arrangements for restricting occupational exposure should be applied to the source of radiation and to features of the workplace. The use of personal protective equipment should in general be supplementary to these more fundamental provisions. The emphasis should thus be on intrinsic safety in the workplace and only secondarily on protection that depends on the worker's own actions," such as the ingestion of KI or use of respiratory equipment.

As a design criterion, GDC 19 does not supplant the radiation protection standards of 10 CFR Part 20, which treat the radiation exposure of control room operators as occupational exposure.

> *The petitioner recommended that as an alternative to the total removal of dose guidelines from the regulations, most of his concerns could be resolved if the dose criteria were based solely on the whole body dose from noble gases. The NRC does not agree with the proposition that the dose criteria should be based solely on the whole body dose from noble gases. The control room dose criterion of 5 rem whole body or its equivalent to any organ imposes two requirements on licensees: satisfaction of the whole body dose criterion, which is generally dominated by the dose from noble gases; and satisfaction of the organ-specific dose guidelines, which are generally dominated by the thyroid dose from the inhalation of iodine.*

In most cases, demonstrating compliance with thyroid dose guidelines poses a significantly greater challenge to licensees than does compliance with the whole body dose criterion.

The 1999 amendment to 10 CFR 50.67, revised the control room dose limit to allow licensees to show compliance with either the existing limits, using the traditional Technical Information Document (TID)-14844 source term assumptions, or a revised single control room dose criterion of 5 rem TEDE[1], if the licensee adopts the AST. With the ability to reassess a maximum credible radiological release using the AST, many licensees have shown compliance with the § 50.67 single control room dose criterion of 5 rem TEDE. Licensees have accomplished this while achieving an enhanced degree of operational flexibility not realized using the traditional TID-14844 source term with the associated whole body dose criterion and organ dose guidelines. Because compliance with § 50.67 is demonstrated by calculating the TEDE, the relative contribution of the thyroid dose to the demonstration of compliance with the control room criterion has been substantially and appropriately reduced.

In addition, many licensees that continue to use the traditional TID-14844 source term have incorporated the guidance in Regulatory Guide (RG) 1.195, "Methods and Assumptions for Evaluating Radiological Consequences for Design-Basis Accidents at Light-Water Nuclear Power Reactors," to achieve operational flexibility. Following the guidance in RG 1.195, licensees are able to evaluate control room habitability using a 50 rem thyroid dose guideline. This represents a significant relaxation from the 30 rem thyroid dose guideline that was incorporated into previous guidance documents.

(2019) PRM-50-122, RE-SUBMITTAL - 10 CFR 2.802 PETITION FOR RULEMAKING ACCIDENT DOSE CRITERIA

Problem Description:

The U.S. Nuclear Regulatory Commission's (NRC's) design basis accident (DBA) dose criteria and the resulting design of accident mitigation systems could be perceived to emphasize protection of the control room operator over protection of the public. The control room criterion restricts the calculated 30-day accident dose to the annual occupational limit of five rem while the off-site dose criteria allows for a calculated dose of 25 rem in two hours. The off-site dose criteria were derived from the siting practices of the earliest reactors and are not reflective of current health physics knowledge or modern plant construction. As a result, the design of accident mitigation systems may not be optimized in the best interest of NRC's mission of protecting public health and safety. The control room accident dose criterion has proven to be challenging to demonstrate with most plants having very little margin to the regulation.

Proposed Solution:

The proposed voluntary rule would allow licensees to adopt revised accident dose criteria that will; (1) be reflective of modern health physics recommendations and modern plant designs, (2) provide a better balance between protection of the control room operator and protection of the public, and (3) relieve the unnecessary regulatory burden associated with meeting the current control room dose criterion.

The attached petition includes the history of the current dose criteria, proposed changes to § 50.67 Accident source term and General Deign Criterion 19, corresponding revisions to Regulatory Guide 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, as well as other supporting information.

The petitioner has attempted to gain support from NRC staff to initiate rulemaking through internal processes for over ten years without success. The referenced dose criteria are codified in NRC regulations. Since internal processes such as the Non-Concurrence Process and the Differing Profession Opinion process are not applicable to concerns with regulations, the petitioner reluctantly submits the attached § 2.802 Petition for rulemaking as an individual.

PURPOSE:

The U.S. Nuclear Regulatory Commission's (NRC's) design basis accident (DBA) dose criteria and the resulting design of accident mitigation systems could be perceived to emphasize protection of the control room operator over protection of the public. The control room criterion restricts the calculated 30-day accident dose to the annual occupational limit of five rem while the off-site dose criteria allows for a calculated dose of 25 rem in two hours. DBA dose criteria should not be viewed as representing actual doses received by individuals but rather as figures of merit which have a direct impact on the design of structures, systems and components (SSCs) important to safety. The off-site dose criteria were derived from the siting practices of the earliest reactors and are not reflective of current health physics knowledge or modern plant construction. As a result, the design of accident mitigation systems may not be optimized in the best interest of NRC's mission of protecting public health and safety. The control room accident dose criterion has proven to be challenging to demonstrate with many plants having very little margin to the regulation.

The purpose of this petition is to identify concerns with current DBA dose criteria and to recommend a proposed voluntary rule allowing licensees to adopt revised accident dose acceptance criteria that will; (1) be reflective of modern health physics recommendations and modern plant designs, (2) provide a better balance between protection of the control room operator and protection of the public, and (3) relieve the unnecessary regulatory burden associated with meeting the current control room dose criterion.

SUMMARY:

During the 1950s, applicants for reactor construction permits submitted Hazards Summary Reports to the Atomic Energy Commission (AEC) describing the potential dose consequences from what was considered the "maximum credible accident."[1] These evaluations contained wide variations in both the assumed source terms as well as the proposed dose acceptance criteria. In response to the recognition that more definitive siting criteria was needed, the AEC developed a procedural methodology to define reactor siting criteria that was generally consistent with the siting practices in effect at the time. There was a concern within the AEC that it was premature to codify these criteria so early in the development of the nuclear power industry. Notwithstanding this concern, in 1962, the AEC published 10 CFR Part 100, "Reactor Site Criteria", specifying dose acceptance criteria of 25 rem whole body and 300 rem thyroid for a 2 hour period at the Exclusion Area Boundary (EAB) and for the accident duration at the outer boundary of the Low Population Zone (LPZ).

The stated objective of the reactor siting criteria was to avoid serious injury to individuals if an unlikely, but still credible, accident should occur. Both the 25 rem criterion and the concept of an exclusion area addressed the potential for extreme radiological hazards that would exist if a fuel melt source term was released into an unshielded containment[2]. The regulation states that the 25 rem whole body corresponds to the once-in-a-lifetime accidental or emergency dose for radiation workers which according to 1959 national council on radiation protection (NCRP) recommendations may be disregarded in the determination of their radiation exposure status[3]. There is no analogous citation for the 300 rem thyroid dose criterion which was not the dose equivalent to 25 rem whole body. Radiation protection standards at the time would have suggested a 6:1 ratio of thyroid to whole body dose

(resulting in 150 rem) so the 300 rem was somewhat arbitrary. The codification of site criteria fulfilled the need to reduce the subjective nature of judging site suitability while providing a methodology that did not conflict with siting decisions already made by the AEC. The regulation was intended to be an interim measure until the state-of-the-art allowed for more definitive standards to be developed.

In 1971 Appendix A, "General Design Criteria for Nuclear Power Plants," was added to 10 CFR Part 50. General Design Criterion 19 (GDC-19) specified that adequate protection shall be provided to permit access and occupancy of the control room for the duration of an accident without exceeding a radiation exposure of 5 rem whole body or its equivalent to any part of the body. The originally stated objective for the 5 rem control room accident dose criterion is not readably traceable however the NRC staff believes that the primary objective of the criterion was to provide a safe, comfortable environment that would enable the control room operators to focus attention on accident mitigation. The numerical value chosen fulfilled this objective however the alignment of the control room accident dose criterion with the annual limit for occupational dose has been an ongoing challenge for licensees. The 5 rem control room dose criterion is limiting for many licensees and this raises the question regarding whether a slightly higher value could still satisfy the objective of providing a comfortable environment for the operators while reducing regulatory burden by increasing the small margin many licensees have relative to the current acceptance criterion.

In the late 1970s there were concerns within the NRC that siting practices were not providing enough emphasis on site isolation as an important contributor to defense-in-depth because engineered safety feature (ESF) systems could be designed to make almost any site acceptable from an accident dose calculation point of view. In August 1978, the NRC directed the staff to develop a general policy statement on nuclear power reactor siting which resulted in NUREG- 0625, "Report of the Siting Policy Task Force," recommending that fixed distances should be required for the EAB and the LPZ in lieu of dose consequence analyses. After numerous comments objecting to a proposed rule (57 FR 47802), which was based on NUREG-0625 recommendations, the commission decided to retain source term and dose calculations by relocating a new single dose criterion based on total effective dose equivalent (TEDE) in 10 CFR 50.34 (61 FR 65157 December 11, 1996).

The new TEDE criterion is applicable to all new reactors and existing reactors that choose to adopt the alternative source term (AST) methodology. Depending on the contribution to TEDE dose from iodine in the released source term, the 25 rem TEDE criterion allows for the associated thyroid dose to substantially exceed the previously controlling 300 rem thyroid limitation. Therefore, new reactors are being sited with a less restrictive dose criterion than the earliest reactors.

Modern health physics recommendations suggest that a dose of 25 rem is difficult to justify as adequately fulfilling the objective of not causing serious harm especially when considering the most dose-sensitive members of the public. The same health physics recommendations indicate that the 5 rem control room dose criterion may be overly restrictive.

Therefore, it is recommended that a uniform design basis accident dose criterion of 10 rem TEDE for the control room, EAB, and LPZ boundary be available to licensees on a voluntary basis. Adoption of this voluntary rule would result in a less restrictive control room dose criterion while significantly strengthening the offsite dose criterion. This voluntary change would provide various benefits in that:

(1) it is technically defensible based on modern health physics guidance indicating that an increased cancer risk is not expected for exposures below 10 rem;

(2) it would avoid the poor optics of allowing a higher design basis dose criterion for members of the public (including the most dose-sensitive groups such as children and pregnant women) than for highly trained nuclear professionals occupying the control room;

(3) it would motivate licensees to provide greater emphasis on offsite dose reduction commensurate with NRC's mission to protect public health and safety; and

(4) it would reduce the regulatory burden required to demonstrate the unnecessarily restrictive 5 rem control room dose criterion.

A significant number of plants would be able to meet a uniform 10 rem TEDE dose criterion without making any changes to their dose consequence analyses. Those plants whose existing DBA dose analyses would be challenged by a 10 rem TEDE dose criterion may be able to increase the credit taken for mitigation systems designed to limit releases to the environment while achieving an increased margin in their control room dose analyses. However, no action on the part of any licensees would be required since the proposed rule presented herein would be available for adoption on a voluntary basis.

NUREG-1433 Standard Technical Specifications — General Electric Plants (BWR/4): Specifications (Revision 4, Volumes 1 and 2)

SR 3.6.1.3.13: "Verify leakage rate through each MSIV is ≤ [11.5] scfh when tested at ≥ [28.8] psig."

SR 3.6.1.3.13 The analyses in References 1 and 8 are based on leakage that is less than the specified leakage rate. Leakage through each MSIV must be ≤ [11.5] scfh when tested at ≥ Pt ([28.8] psig). A Note is added to this SR which states that these valves are only required to meet this leakage limit in MODES 1, 2, and 3. In the other conditions, the Reactor Coolant System is not pressurized and specific primary containment leakage limits are not required. This ensures that MSIV leakage is properly accounted for in determining the overall primary containment

leakage rate. The Frequency is required by the Primary Containment Leakage Rate Testing Program.

INSPECTION MANUAL CHAPTER 0308 ATTACHMENT 1 TECHNICAL BASIS FOR PERFORMANCE INDICATORS Effective Date: 01/01/2021

Performance Indicator: Reactor Coolant System Leakage

Cornerstone: Barrier Integrity

Objective: This indicator monitors the integrity of the RCS pressure boundary, the second of the three barriers to prevent the release of fission products. It measures RCS Identified Leakage as a percentage of the technical specification allowable Identified Leakage to provide an indication of RCS integrity.