

the SBO rule was issued, all plants have established an EDG reliability program that has improved EDG reliability. A report shows that only 3 of 102 operating plants have a unit average EDG reliability less than 0.95 and above 0.90 considering actual performance on demand, and maintenance (and testing) out of service (MOOS) with the reactor at power.”

Therefore, the NRC staff has concluded that requirements for hydrogen and oxygen monitors can be removed from technical specifications. The basis for this conclusion is:

1. These monitors do not meet the criteria of 10 CFR 50.36,
2. The amended 10 CFR 50.44 requires hydrogen and oxygen monitors to be maintained reliable and functional, and
3. The regulatory precedents set by the treatment of other equipment for severe accidents required by 10 CFR 50.62 and 50.63.

F. Combustible Gas Control Requirements for Future Applicants

Section 50.44(c) of the final rule sets forth combustible gas control requirements for all future water-cooled nuclear power reactor designs with characteristics (e.g. type and quantity of cladding materials) such that the potential for production of combustible gases is comparable to currently-licensed light-water reactor designs. The NRC's requirements for future reactors previously specified in § 50.34(f)(2)(ix) have been reworded for conciseness but without material change and relocated to § 50.44(c)(2) to consolidate the combustible gas control requirements in § 50.44 for easier reference. This subparagraph requires a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel clad metal-water reaction and must be capable of precluding uniformly distributed concentrations of hydrogen from exceeding 10 percent (by volume). If these conditions cannot be satisfied, an inerted atmosphere must be provided within the containment. The requirements specified in amended § 50.44(c)(2) are applicable to future water-cooled reactors with the same potential for the production of combustible gas as currently-licensed light-water reactor designs and are consistent with the criteria currently contained in § 50.34(f)(2)(ix) to preclude local concentrations of hydrogen collecting in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate accident mitigating features. Additional advantages of providing

hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) include the lessening of pressure and temperature loadings on the containment and essential equipment. These requirements reflect the Commission's expectation that future designs will achieve a higher standard of severe accident performance (50 FR 32138; August 8, 1985).

Section 50.44(d) applies to non-water-cooled reactors and water-cooled reactors that have different characteristics regarding the production of combustible gases from current light-water reactors. Because the specific details of the designs and construction materials used in such future reactors cannot now be known, paragraph (d) specifies a general performance-based requirement that future applicants submit information to the NRC indicating how the safety impacts of combustible gases generated during design-basis and significant beyond design-basis accidents are addressed to ensure adequate protection of public health and safety and common defense and security. This information must be based in part upon a design-specific probabilistic risk assessment. The Commission has endorsed the use of PRAs as a tool in regulatory decisionmaking, see *Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement* (60 FR 42622, August 16, 1995), and is currently using PRAs as one element in evaluating proposed changes to licensing bases for currently licensed nuclear power plants, see Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisionmaking: General Guidance* (July 1998) and Standard Review Plan, Chapter 19, “Use of Probabilistic Risk Assessment in Plant-Specific, Risk Informed Decisionmaking: General Guidance,” NUREG-0800 (July 1998). The use of PRA methodologies in determining whether severe accidents involving combustible gas must be addressed by future non-water-cooled reactor designs (and water-cooled designs which have different combustible gas generation characteristics as compared with the current fleet of light-water-cooled reactors) is a logical extension of the NRC's efforts to expand the use of PRAs in regulatory decisionmaking.

At this time, the NRC is not able to set forth a detailed description of, or specific criteria for defining a “significant” beyond design-basis accident for these future reactor designs, because the fuel and vessel design, cladding material, coolant type, and containment strategy for these reactor

designs are unknown at the time of this final rulemaking. Based in part upon the design-specific PRA, the NRC will determine: (i) What type of accident is considered “significant” for each future reactor design, (ii) whether combustible gas control measures are necessary, and if so, (iii) whether the combustible gas control measures proposed for each design provide adequate protection to public health and safety and common defense and security. Although it is impossible at this time to provide a detailed description or criteria for determining what constitutes a “significant” beyond design-basis accident for the future reactors that are subject to this provision, the NRC nonetheless believes that the concept of “significant” with respect to severe accidents has regulatory precedent which will guide the NRC staff's evaluation of the PRA information for future plants. Section 50.34(f)(2)(ix) of the NRC's current regulations already defines what is in essence the significant beyond design-basis accident which future reactor designs comparable to current light-water reactor designs must be capable of addressing, viz., an accident comparable to a degraded core accident at a current light-water reactor in which a metal-water reaction occurs involving 100 percent of the fuel cladding surrounding the active fuel region (excluding the cladding surrounding the plenum volume). With respect to other “beyond design-basis” accidents, the Commission has addressed anticipated transients without scram (ATWS), and station blackout, which are currently regarded as “beyond design-basis accidents.” The nuclear power industry, at the behest of the NRC, has developed severe accident management guidelines to provide for a systematized approach for responding to severe accidents during operations. Finally, the Commission has required all nuclear power plant licensees to implement emergency preparedness planning to address the potential for offsite releases of radiation in excess of 10 CFR Part 100 limits. A careful review of these regulatory efforts discloses a common thread: regulatory actions addressing “beyond design-basis” accidents have generally been determined based upon a consideration of probability of the accident, together with consideration of the potential scope and seriousness of the health and property value impacts to the general public. Thus, it is possible to set forth a high-level conceptual description of a “significant” beyond design-basis accident involving combustible gas for which the