Samuel Miranda Comments (received 11/14/18)

Here are the same comments, minus a couple of typos. Sorry about that.

Here are my comments regarding the Seabrook Station License Renewal Application.

The ACRS Subcommittee on Plant License Renewal is asked consider the following comment with respect to the aging management program for systems, structures, and components that are credited for the renewal of Seabrook's operating license:

According to Seabrook's FSAR (Chapter 15.0.8, "Plant Systems and Components Available for Mitigation of Accident Effects", ADAMS No. ML17310A427), "In determining which systems are necessary to mitigate the effects of these postulated events, the classification system of ANSI N18.2-1973 is utilized." Chapter 15.4.3, "Rod Cluster Control Assembly Misoperation (System Malfunction or Operator Error)" discusses the single RCCA withdrawal, and indicates that, "consistent with the philosophy and format of ANSI N18.2, the event is classified as a Condition III event. By definition, "Condition III occurrences include incidents, any one o f which may occur during the lifetime of a particular plant," and "shall not cause more than a small fraction of fuel elements in the reactor to be damaged..."

It can be expected that an extension of Seabrook's operating lifetime by 20 years (i.e., by 50%) will cause a proportional increase in the frequency, on average, of Condition III Infrequent Incidents (i.e., by 50%). It seems that a license renewal could significantly increase the probability of a previously evaluated (Condition III) accident. Could Seabrook claim that there is no significant hazard associated with the proposed license renewal?

Is Seabrook proposing to make a change (e.g., in plant design or operation) in order to reduce the frequency of occurrence of Condition III events to the value that is specified in the original operating license? That is, will there a change that will decrease the frequency of occurrence of Condition III events from ≤ 1/40 reactor-years to ≤ 1/60 reactor-years?

The staff's SER for the Seabrook LRA, Chapter 4.5, "Concrete Containment Tendon Pre-stress" (ADAMS No, ML18354A294), indicates, "The LRA also states that the 40-year values were increased (linearly) by 50 percent to derive the projected values for 60 years of operation, and that the 60-year projections continue to satisfy the criteria of ASME Code Section III Article NE-3221.5(d)." Can a similar statement of extrapolation be made for the Condition III events that are reported in FSAR Chapter 15?

Probabilistic risk assessment (PRA) arguments could well dismiss the occurrence of infrequent incidents, as highly unlikely; but the use of PRA would be inappropriate in this application. This is because 10 CFR §54 requires that plants maintain their current, deterministic licensing bases during the extended terms of operation that are authorized by their renewed licenses. Consider that an even less likely class of events, anticipated transients without scram (ATWS) is specifically listed in the scope of 10 CFR §54. The definition of scope, as defined in 10 CFR §54.4, includes, "(a) Plant systems, structures, and components within the scope of this part are … (3) All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for …. Anticipated transients without scram (10 CFR 50.62)." ATWS events are not likely to occur, and not included in plant design bases.

This is supported by the Statement of Consideration, "The Commission reaffirms its previous conclusion (see 56 FR 64943 - 64956) that PRA techniques are most valuable when they focus the traditional, deterministic-based regulations and support the defense-in depth philosophy. In this regard, PRA methods and techniques would focus regulations and programs on those items most important to safety by eliminating unnecessary conservatism or by supporting additional regulatory requirements. PRA insights would be used to more clearly define a proper safety focus, which may be narrower or may be broader. In any case, PRA will not be used to justify poor performance in aging management or to reduce regulatory or programmatic requirements to the extent that the implementation of the regulation or program is no longer adequate to credit for monitoring or identifying the effects of aging." --- FR 22468, Vol. 60, No. 88 (May 8, 1995)

10 CFR §54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*, "governs the issuance of renewed operating licenses for nuclear power plants." So, is the renewal of an operating license the same as the issuance of a renewed operating license? If yes, then why is 10 CFR §54 required? Would it not be simpler, and less confusing, to issue a license amendment, under 10 CFR §50, which would extend the license expiration date, and record a license commitment (or condition) to establish and implement an acceptable aging management program? Then the new expiration date would be specified in a license amendment that converts a 40-year license into a 60-year license. Approval of the license renewal, as an amendment, would also be subject to the requirements of 10 CFR §50.92, *Issuance of amendment,* which addresses, among other things, the question of whether the operation of the facility, in accordance with the proposed amendment, would cause a significant increase (e.g., 50%) in the probability of an accident (e.g., an infrequent incident) previously evaluated. In this way, (1) the CLB is maintained, (2) there is no doubt as to whether all amendments and commitments that were made for a 40-year license also apply to a 60-year license, and (3) the license renewal is accomplished by amendment to an existing license, consistent with all other major changes (e.g. power upratings); not by issuing a "renewed license".