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Sent: July-07-16 2:52 PM
To: Consultation (CNSC/CCSN)
Cc: Nalasco, Adrian (ENERGY); Anam, Zobair (ENERGY); Thiru, Dhil (ENERGY)
Subject: Comments on CNSC discussion paper DIS-16-04,Small Modular Reactors: Regulatory Strategy, Approaches and Challenges

Please find below my comments on the CNSC Discussion Paper DIS-16-04 " Small Modular Reactors: Regulatory Strategy, Approaches and Challenges":

From Section 2.4

For the topic of "Licensing approach for a new demonstration reactor", is there a need for additional clarification or information beyond that found in RD/GD-369, *Licence Application Guide: Licence to Construct a Nuclear Power Plant*?

If yes, what needs to be clarified or added?

Comment:

RD/GD-369, License Application Guide: ***License to Construct a Nuclear Power Plant*** *is a CNSC regulatory guide intended to establish the scope of CNSC staff's review of an application to construct a nuclear power plant near design completion, with three integrated levels of objectives, resulting in an overall assessment of the adequacy of a license application.*

A new demonstration reactor has many design uncertainties because it is still in early design prototype stage. Many aspects of the requirements in RD/GD-369, for example, Section 9 Operational Aspects and Section 10 Operational Limits and Conditions can be more defined after the demonstration plant has been built and tested. Therefore it will be difficult for the licensee of a demonstration reactor to provide all the required documentation in one license application as per Appendix B: Relevant Requirements in RD/GD-369.

Therefore, with respect to the licensing of a demonstration reactor, it will be helpful for Appendix B of RD/GD-369 to provide guidance, for instance:

1. *To prioritize the sequence of submission of Document Section in a manner commensurate with the CNSC risk-informed regulation, that is appropriate for addressing safety risks for demonstration reactor in prototype design stage, and;*
2. *To structure a graded approach perhaps using a safety significant classification scheme (references: REGDOC 2.5.2, Clause 7.1 Safety classification of structures, systems and components; IAEA SSG-30: Safety Classification of Structures, Systems and Components in Nuclear Power Plant), thus enabling the applicant to apply graded approach to establish the stringency of design measures, safety analyses and provisions for conduct of their activities commensurate with the level of risk posed by the reactor facility (reference : 5th paragraph of Executive Summary of this Discussion Paper).*

With respect to addressing uncertainties introduced by the application of integrated multiple novel features in a demonstration facility, are requirements regarding the scope and adequacy of supporting information sufficiently clear?

Comment:

The scope and requirements regarding the scope and adequacy of supporting information are not sufficiently clear.

The scope and adequacy of supporting information as per RD/GD-369 depend on the quality of the deterministic and probabilistic safety analyses (DSA and PSA) of the new reactor design to address the uncertainties introduced by the application of integrated multiple novel features in a demonstration facility.

The main challenges to developing a quality PSA for a demonstration reactor are related to the intrinsic difficulties to ensure the representativeness and the quality of the model for a demonstration reactor in preoperational phase. The Authority Having Jurisdiction (AHJ) may be in a position to make decisions using a PSA which may not exactly reflect the future as-built as-operated plant. In addition, the technical challenges on AHJ may include the need to address very different systems and phenomenology than water-cooled reactor technology, the potential unavailability of important reliability and experimental data, the potential unavailability of knowledge on new key phenomenon, and the potential unavailability of accident analysis models.

In light of the intrinsic difficulties as mentioned, the result of implementing RD/GD-360 as-is will likely result in a "stop-and-go" situation for the licensing "clock", waiting for "quality" documentation to be submitted by the licensee. Paradoxically, some of the quality licensing documentation e.g. experimental data; knowledge of new key phenomena associated with novel features, etc. can only be made available after the demonstration reactor is built and tested.

Henceforth, it seems that RD/GD-369 needs a front-end piece to address the uncertainties early on in the licensing process introduced by the application of integrated multiple novel features.

In an effort to develop a more detailed understanding of safety related design vulnerabilities, and the resulting contributions to risk, the GEN IV Forum (GIF) has developed an Integrated Safety Assessment Methodology (ISAM) for Generation IV Nuclear Systems.
https://www.gen-4.org/gif/upload/docs/application/pdf/2013-09/gif_rsgw_2010_2_isamrev1_finalforeg17june2011.pdf

The ISAM safety assessment methodology is technology neutral, regardless of the design difference in coolant, fuel and inherent safety novel features. The advantage of this Methodology is that it can complement existing CNSC licensing framework (vis-à-

vis RD/GD-369), in a manner that provides early informed risked based understanding of safety vulnerabilities, thus allowing AHJ to identify new safety provisions or design improvements to be implemented during the licensing process.

ISAM involves five analytical tools: 1. Qualitative Safety Features Review (QSR) 2. Phenomena Identification Ranking Table (PIRT) 3. Objective Provision Tree (OPT) – provisions that assure the implementation of Defense-in-Depth 4. Probabilistic Safety Analysis (PSA) 5. Deterministic and Phenomenological Analysis (DPA) , with each tool intended to address specific kinds of safety-related issues at different design phases. The purpose of the ISAM tool set is to inform the design process and to help structure inputs that will eventually make its way into the quality PSA discussed earlier. Finally regulatory requirements are defined through the detailed safety analysis using CNSC regulatory licensing framework by integrating ISAM results.

As a reference, ISAM has been developed as Proof-of-Concept for:

1. *Korean Fusion DEMO Plant (K-DEMO): http://www.kns.org/kns_files/kns/file/13S-07B-11A-%BF%C0%B0%E8%B9%CE.pdf*
2. *Safety Approach of Gen IV System – Application to Sodium Cooled Fast Reactor (SFR): https://www.iaea.org/INPRO/cooperation/IAEA-GIF_WS_on_SFRs/8_GLF-Safety_Approach_of_Gen-IV_systems_Application_to_SFR_Short_V1.pdf*
3. *Lead- Cooled Fast Reactor (LFR) Risk and Safety Assessment : https://www.gen-4.org/gif/upload/docs/application/pdf/2014-11/rswg_lfr_white_paper_final_8.0.pdf*
4. *Molten Salt Reactor (Section A.IV – Molten Salt Reactor Safety Analysis) : http://cordis.europa.eu/docs/results/249/249696/final1-final-report-f.pdf*
5. *High temperature Gas Cooled Reactor : Use of Objective Provisions Tree (OPT) http://www-pub.iaea.org/MTCD/publications/PDF/te_1366_web.pdf*

In summary, ISAM seems to offer a good front end piece for RD/GD-369 due to the following desirable characteristics:

- *Consists of, or is largely based on, existing tools that are widely accepted for their validity. Minimizes need for development of new techniques.*
- *Practical and flexible - allows for graded approach to technical issues of varying complexity and importance. Offers analysis tools tailored to appropriate stage of design.*
- *Identifies vulnerabilities and relative contributions to risk.*
- *Allows for explicit consideration and characterization of uncertainty.*
- *Supports integration of multidisciplinary inputs.*
- *Combines probabilistic and deterministic perspectives.*
- *Consistent with international GIF Risk and Safety Working Group (RSWG) safety philosophy, Proliferation Resistance and Physical Protection methodology, and other relevant work (US NRC NUREG-1860, IAEA TECDOC-1570, etc).*

What, if any, requirements need to be revisited to address activities involving demonstration reactors? For example, are additional requirements or guidance needed to address operational restrictions if the facility is being used to gather operating experience that would be normally be needed for commercial facility licences?

Comment:

As discussed above, the ISAM tools facilitate early informed risked based understanding of safety vulnerabilities, thus allowing AHJ to identify new safety provisions or design improvements to be implemented during the licensing process.

If the facility is being used to gather operating experience that would be normally be needed for commercial facility licences, additional requirements or guidance are needed to address operational restrictions.

As an example, if the demonstration plant is built with prototype design criteria to demonstrate low power operation, risk analysis or (QSR) is developed using low power operation design criteria as inputs, PIRT is performed to identify a spectrum of safety related phenomena or scenarios on the basis of their safety significance on low power operation. OPT(DiD) is then developed , focusing on identifying design provisions to prevent, control, mitigate the consequence of the low power PIRT analysis, followed by PSA and DPA for low power operation. These ISAM results provide inputs for licensing conditions only for low power demonstration.

Once the Demonstrator reactor design can be scaled up for higher power operation, the ISAM process is repeated to provide analysis results for all plant states using QSR, PIRT, OPT, PSA and DAP for higher power and low power operations. Similarly these ISAM results provide inputs for licensing conditions for higher and low power demonstration. Safety Requirements deemed to be satisfied on low power operation would be revisited, if the high power ISAM analysis discovers new issues of such requirements.

<u>From Section 2.8</u>

For the topic of “deterministic/probabilistic safety analyses”, are the regulatory requirements and guidance clear for the kinds of alternatives that might be proposed for Deterministic/probabilistic safety analyses for SMR facilities?

Do the existing requirements permit the establishment of a suitable level of probabilistic safety analysis for different novel designs?
Does enough information currently exist to apply probabilistic safety analysis to novel designs?

Comments:

As discussed in previous Section 2.4, it is suggested that the DSA and PSA are augmented with ISAM tools, which are used to inform the design process and to help structure inputs that will eventually make its way into the quality PSA discussed earlier. Finally regulatory requirements are defined through the detailed safety analysis using CNSC regulatory licensing framework integrating ISAM results as front end.

From Section 2.9

For the topic of “defence in depth and mitigation of accidents”, given some of the novel safety approaches that vendors are proposing, are the existing requirements and guidance around defence in depth adequately clear for prevention and mitigation of accidents? Consider this question with particular attention to the following topics and combinations thereof:

- application of inherent and/or passive safety features
- application of alternative instrumentation and control strategies (e.g., remote monitoring and intervention of a fully-automated facility)
- non-water cooled technologies
- transportable sealed and factory fueled SMRs (see section 2.11)
- facilities proposed to be located in highly remote regions

Comment:

In IAEA TECDOC 1366 http://www-pub.iaea.org/MTCD/publications/PDF/te_1366_web.pdf, there is a good case study presentation on High Temperature Gas Cooled Reactor (HTGR), given its novel passive safety feature, to use the approach of Objective Provision Trees (OPT) to develop an extended interpretation of the concept of defence in depth and its linkages with the general safety objectives and fundamental safety functions as set out in IAEA NS-R.1

The current DiD requirements in REGDOC 2.5.2 is adequate for water cooled reactor technology.

The implementation of defence in depth (DiD.) for HTGR differs from that for the traditional water-cooled reactor strategy to achieve effective defence against radiological hazards. The safety
of HTGR relies strongly on inherent features, with the confinement of radionuclides being accomplished with minimal or no reliance on active systems or operator actions.

The use of Objective Provision Tree (OPT) approach as illustrated in TECDOC 1366 appears to provide defensible DiD implemenation for advanced reactor with novel features.

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