CMD 22-H8.14

File / dossier : 6.01.07
Date: 2022-03-12
Edocs: 6750987

Oral Presentation

Written submission from David J. Winfield

In the Matter of the

SRB Technologies (Canada) Inc.

Application for the renewal of the licence for SRBT Facility

Commission Public Hearing

April 27 and/or 28, 2022

Exposé oral

Mémoire de David J. Winfield

À l'égard de

SRB Technologies (Canada) Inc.

Demande de renouvellement de permis pour l'installation de SRBT

Audience publique de la Commission

27 et/ou 28 avril 2022

Comments on Application for the April 27-28, 2022

Renewal of the Licence for the SRBT Facility

Reference: CMD 2022 H-08

D. J. Winfield

Table of Contents

Executive Summary

SRBT is requesting, in CMD 22-H-08, a licence renewal period of 15 years. My opinion for the Commission is that such a period, while recognized as lengthy, is warranted and can currently be justified.

The main bases for this conclusion are (i) that no significant activity changes to the current production processes are likely in the next 15 years (ii) that the organizational structure and staff experience level is well established and stable and (iii) from observations made during a facility tour for the purpose of this intervention. Additionally, the current status of supporting safety documentation, annual reporting, and SRBT's demonstrated improved performance since the 2015 licence renewal, indicates a continued high-level commitment to operate the facility safely and compliantly.

Significant resources spent on more frequent licence renewal preparation activities and also ACR report preparation could however be more usefully expended by CNSC and SRBT focusing on the key technical safety issue; minimizing tritium releases, though design and operations. With this in mind, recommendations are made, relevant to both SRBT and CNSC staff, to improve the Annual Compliance Report (ACR) and Safety Analysis Report (SAR) content. This intervenor's impressionism is that an excessive increase in size over time for both these documents has developed, at the expense of technical quality, ease of readability and usefulness for public review. While a 15-year licence period limits formal opportunities for public intervenors, the CNSC still has authority under the Nuclear Safety and Control Act (NSCA) to control any aspect of a current licence. The public would still not be precluded from providing unsolicited comments to the CNSC and SRBT at any time, with no artificial review time constraints, albeit not in a formal public forum.

A relatively new potential external hazard from a nearby liquid propane distribution facility was identified in the 2017 SAR. From the limited information available to SRBT for the SAR, the intervenor was unable to judge whether the risk from this facility should be discounted and recommends CNSC staff obtain more information from the regulator of this facility.

Both tritium supply and product demand for SRBT's products would seem assured for decades, with effectively a closed loop from production, recycling to waste storage well established with any major changes unlikely. The potential for market impact from future use of photo-voltaic sources, noted by some intervenors for the 2015 SRBT licence hearings, does not appear to be occurring.

1. Review

I have reviewed submissions 22 H-08-1, 22 H-08, the historical ACR's and the SAR in sufficient detail to provide some general comments for SRBT, CNSC staff and the Commission from a public outsider's perspective. The comments are intended not only to provide input for the re-licencing submission decision, but with intent to encourage future safety-relevant documentation improvement. Significantly the intervenor commends to the Commission that the SAR, the ACRs, Reportable Events, Environment Risk Assessment, Benchmarking and other technical documentation have all been made easily and publically available on SRBT's web site. Particularly noteworthy for public disclosure is the absence of proprietary technology claims, often used needlessly by licensees of much larger facilities, both private and public to obscure even the most basic public reviews of licensee facilities. Such public disclosure for a small production facility, particularly for the SAR, the EA and reportable events, in what appears to be essentially entirely unredacted content, is thus noteworthy to mention to the Commission. With much detailed information this openness thus enables public opportunity for many questions and comments. Such openness can be costly and resource intensive to address. On the other hand openness and transparency offers opportunity for potential improvement in all aspects of a licensee's activities, in addition to potentially improving public support and trust.

2. Annual Compliance Reports Review

The intervenor finds the current ACR content is challenging to review because of unnecessarily excessive content requirements, over 300 pages for an annual summary. From 2000 to 2020 the ACR content increased from 6 pages to over 300 pages. This length now exceeds annually the entire content of many SARs for significantly larger and much higher risk nuclear reactor facilities that the intervenor is aware of. A graded risk-informed regulatory approach does not seem apparent. This is despite SRBT using a single radioisotope in a relatively simple overall production process, recognizing that individual process activities are complex and precise, but on very small physical scales. It is also expected that individual production processes will not differ in the foreseeable future. Issues of major maintenance, major obsolescence, ageing management, equipment qualification and difficult equipment inspection regimes are not factors of any significant safety concern, by comparison with much larger complex reactor facilities.

The prescriptive requirement of fourteen Safety Control Areas (SCAs) in the SRBT ACRs since 2011 seems to be a slightly modified version of fourteen safety factors used in the IAEA SSG-25 nuclear power plant and the, essentially identical, IAEA SRS-99 research reactor documents, These safety factors were introduced by the IAEA as topics to be assessed for reactors for licence renewals and / or for long-term Periodic Safety Reviews (PSRs) of 10+ years periodicity. The IAEA safety factors were not designed or intended to be used as content for power or research reactor ACRs, due to the substantial resources and time needed for a meaningful review of, in the main part, issues of longer time concern[1]. Current international experience even for small research reactors, < 10 MW, is that the overall PSR process for fourteen safety factors, itself typically takes ≈2-3 years.

The prescriptive SCA approach might provide some overall regulatory format consistency for nuclear facility licensees. Using identical SCAs however for entirely different types of facilities without recognizing specific facility features seems to be a weakness. The IAEA safety factors focus only on power and research reactors. The intervenor also does not see the application of a

[1] Detailed review criteria for the safety factors are provided in Technical Safety Review (TSR) Service Guidelines, Periodic Safety Review, IAEA Working Document, July 2, 2020.

graded approach in the significant-volume example of detail in the SRBT ACRs, compared to NPP licensing with their orders-of-magnitude upscaling in equipment size, radioactive inventory, operational complexity and staffing.

The ACR format used prior to SCA's, seems to this reviewer to have been more performance based, and usefully more suitably focused on safety topics. Apart from excessive content, due to the SCA format, the intervenor believes the assessment of global compliance, that seems to be the regulatory oversight goal[2], is deficient due to inherent flaws in the method. A well-established standardized and auditable methodology of deriving a global assessment (rating) for a defined attribute, in this case safety compliance / performance, using a list of objective and subjective variables, (i.e. the fourteen SCAs), is the Analytic Hierarchy Process (AHP). This does not seem to have been applied in Table F-5 of the oversight report[2] even in a cursory way, as the tick-box approach assumes that each SCA is equally ranked in safety importance for the time period of one year, which is not the case. When assigning relative rankings, SCA's should be independent; they are not, and there are a number of interconnections. Assessing SCA Physical Design for instance might imply judging effort spent on design / safety reviews aimed at revealing hidden design flaws and tracking of design modifications made. The latter for example might more appropriately by dominated by design change control activities, under SCA Operating Performance. For a small production facility such as SRBT with a fixed highly specialized product line, it may well be that the Physical Design SCA is not all relevant in the short time of one year. SCA, Safeguards and Non-Proliferation is itself ranked N/A in the oversight report, but is included in the ACR.

The intervenor concludes that the SCA ratings in the Regulatory Oversight Report for SRBT will give rise to an overall conclusion, not supported, or auditable, by standard decision analysis methodology. For instance giving a satisfactory (SA) rating to each of the thirteen remaining SCA's implies the importance of each SCA to safety is exactly the same, which is not the case. Misleading conclusions from a matrix such as Table F-5, are thus likely[2]. The 2020 discontinuance of the CNSC's previous use of the Fully Satisfactory (FS) rating[3] demonstrates, in my view, a methodology deficiency[4].

The graded approach was first introduced into IAEA standard documentation with IAEA SSG-22, for research reactors. Research reactors were specifically used by the IAEA first, due to the many orders of magnitude variations in reactor power and design complexity, as examples of facilities of a common type where scalability for safety and licensing could logically be applied. Use of a graded approach between entirely different types of facilities but using the identical SCA listing for regulatory compliance is more problematic. A comparison, which maybe useful from an overall risk perspective in the context of using a graded approach might be to compare the SRBT tritium possession licence limit with the tritium content in the moderator system (i.e. in HTO form) of a single CANDU unit and also with the Darlington Tritium Removal Facility tritium limit. For SRBT with its limited quantity of a single isotope the intervenor does not though see the claimed usefulness[5] of grading in a 300-page ACR.

---

2 Regulatory Oversight Report for Uranium and Nuclear Substance Processing Facilities and Research Reactors in Canada: 2020, Table F-5, SCA Ratings, SRB Technologies (Canada), Inc., 2016-2020.

3 Ibid, Section 5.2.

4 CMD H22-H8 Reference A. Risk Ranking

5 http://www.nuclearsafety.gc.ca/eng/pdfs/Presentations/CNSC_Staff/2017/20170810-doug-miller-use-of-graded-approach-eng.pdf

Much detail especially of fixed description requiring infrequent updating and of little direct relevance to annual review could logically be located in other technical reference documents. A focus on OLCs, introduced it seems only in 2015, would provide a major measure of regulatory requirements of a practical and relevant nature and seem to be ample to cover most of the needs of an ACR. Repeating the text of OLC's when there are no exceedances seems unnecessary and leads to information overload for the public reader. Only exceedances, precursor events or systematic trends need to be mentioned in an ACR. Historical trends of emissions, radiation doses are key parameters for inclusion and these are indeed well documented in the ACR. Unnecessarily repeating identical information in different tables e.g. tritium releases, detracts from readability though.

Prior to the use of SCA's, not used internationally by the IAEA for ACRs, the SRBT ACR's were more focused on essential safety issues and much easier to comprehend. The SCAs may provide a convenient approach for reporting consistency between licensees, but this does not address major differences between reactor and non-reactor licensee facilities. My impression is the SCA approach tends to lose focus on key safety features that in any case can be captured almost entirely by facility specific OLC's. For SRBT these are very well defined in the SAR Chapter 10. Root causes of any serious OLC non-compliances / exceedances would be identified by the Reportable Event process. These root causes would then be classified into one or another of the SCA's. An appropriate measure of annual compliance should be dominated by quantifiable impact. The extent of OLC non-compliance will always be facility specific, leading in turn attention to root causes which would invariably be linked into an SCA appropriate for a specifc facility.

3. SAR Review

In the review time available the intervenor reviewed the SAR only generally but not in detail. Four topics are mentioned below. Section 3.1 is recommended to be addressed by CNSC staff in the near term and Sections 3.2 to 3.4 recommended to be addressed for longer term updating of the SAR. Overarching comments would be to aim for improved integration of the analysis presented, provide concise summaries of analysis assumptions, minimise excessive detailed discussion of topics with little relevance to safety and reducing obsolete details. The intervenor recognises such SAR updating requires a significant undertaking.

3.1 External Hazards

Two external event 'worst-case' dose scenarios (C and D) are postulated in Appendix A. The remaining Scenarios A, B, E, F, G are postulated internal events. The external event scenarios do not include any further discussion of the potential external hazard from a propane storage tank BLEVE scenario that was raised in Section 4(l), Proximity of Industrial Facilities. Inclusion of Section 4(l) in the external event scenarios of Appendix A would have been appropriate for completion, as well as to recognize the potential non-radiological risk to staff, building and equipment, of this scenario, which was one of five 'worst-case' scenarios assessed by the safety authority for the propane distribution facility.

Section 4(l) noted that SRBT was about 200 m from the storage tank location, imposed externally by industrial development since about 2017. Concern of a tritium release, when potential fatalities from a BLEVE explosion blast pressure, fireball thermal radiation and flying missile debris might well dominate the risk, would appear to be a much smaller concern. The external risk at the SRBT location from the above-ground 49,000 USWG propane storage tank is quoted, from an MIACC acceptance threshold to be $<10^{-5}$ fatality / year. The intervenor notes this MIACC acceptability criteria is one of individual fatality risk. Considering the number of personnel employed in the TransCanada Corporate Park complex, of which SRBT is located, suggests however that a societal risk acceptability criteria would be more appropriate. The intervenor also notes that an individual fatality risk of >10^{-5} per year is judged unacceptable for new industrial installations in several international jurisdictions, including the Heath and Safety Executive (HSE) of the UK. Not provided in the SAR, nor in any available reference, is a diagram of the predicted individual fatality risk per year contours from the storage tank source, relative to the SRBT location. The intervenor cannot therefore make a judgement of the relevancy, or otherwise, of the quoted <10^{-5} fatality / year as no risk contours are shown. The quoted distance of 200 m has now been more accurately taken as about 250 m. With the presence of a number of additional, albeit much small tanks with possibly unknown additional propane content within the secure area of the large tank, this might result in a distance uncertainty from a potential ignition source to perhaps ±25m.

Bearing in mind the inherent large uncertainty in any probabilistic risk acceptance threshold, such as the MIACC fatality risk criteria, my conclusion is that the brief analysis information provided in Section 4(l) of the SAR (which, as an externally imposed hazard, was not performed or contracted by SRBT) is insufficient to discount any impact on SRBT from this relatively recently imposed external hazard.

The postulated high probability that a warning will be given prior to a potential accident is not substantiated in the SAR. The type of warning initiator is not mentioned, whether it be fire and is automatic (24/7). Typically BLEVEs have extreme consequences and initiating event frequency predictions have very large uncertainties. Accident experience indicates progression, from an initiating fire transitioning into a BLEVE, cannot be well predicted in time, due to widely varying factors. The SRBT staff emergency procedures, suggested immediately following a 'warning', in the SAR might thus be questionable. The SAR claim that in between buildings will attenuate blast pressure and thermal radiation impact provides no reference details. These buildings might though credibly contribute to missile fragment impact damage. Regardless, the in between buildings are not in any case engineered for blast protection, where blast mitigating credit could be claimed. For ground-based storage tanks, engineered BLEVE protection can be provided by an earthen and rock berm / talus, designed to directionally reflect the blast pressure wave.

With probabilistic risk uncertainties being so large with regard to event frequency, the intervenor recommends more information should be obtained from a deterministic consequence analysis of a BLEVE at the storage tank location. The location is basically fairly flat ground. Generic literature analysis is therefore likely to be applicable and adequate to provide a good assessment of the fireball radiation impact and blast overpressure, as a function of distance. Such consequence analysis for these two parameters in particular should be requested and would be expected to deterministically predict consequences for the distance range 200 - 300 m in the direction of SRBT, without large uncertainties. It should be noted that BLEVE impact severities do not increase proportionately with the propane source quantity.

3.2 Worst-case Scenarios

Figure 12 presents potential dose calculations for seven 'worst-case' scenarios. The concept of presenting many hypothetical 'worst' cases is confusing for a public reviewer, e.g. predicted maximum doses for a public receptor at 99 / 100 m vary by factor of six between four of these 'worst cases'. It should be possible to screen these seven cases into one bounding release scenario based on a source term basis, defined by the potentially largest quantity of tritium, thus eliminating much of the extensive dose analysis calculations currently in the SAR and reduce its volume. The SAR is not the location where an overwhelming amount (60 pages) of computer print out with historical input parameters should be archived. There should be no reason why all the Pasquill weather classes need to be analysed; typically the extremes of Pasquill A and F should be adequate. It is not clear if the 99 / 100 m distance has been arbitrarily chosen or actually represents the distance for the peak dose. A dose-distance graph for the two extreme weather conditions would provide clearer information for a reviewer.

Probabilistically defining a 'worst-case' release scenario can be illusory, let alone defining seven of them. Traditionally, but more appropriately named is the design basis accident (DBA). The reviewer should be presented with just a simple bounding case for the facility, defined by a clear summary of realistic upper limits for the many assumptions for the release / dispersion parameters and dose conversion factors used. DBA assumptions used should be conservative but not arbitrarily defined. Examples would be using the facility licensed possession inventory as the source term for atmospheric release; 100% ground level release, Pasquill F weather dispersion and using a 100% HTO DCF for the various dose receptors. This bounding scenario, or some variation upon it, would then, in effect, be the design basis accident release. The IAEA has in recent years introduced 'design extension condition' scenarios for nuclear facilities, formally more sensibly referred to as beyond design basis accident (BDBA) scenarios. With the simplicity of the SRBT facility, dealing with a single isotope, a bounding DBA case could credibly discount any need to consider design extension BDBA cases. The argument might be made that a 100% ground release rather, than an elevated stack release, is a design extension condition. Regardless, a ground level release dose prediction should be made. Also, a well-defined DBA dose estimate would provide a very simple overview and eliminate much analysis. With a maximum DBA dose predicted, the various mitigation factors and assumptions could then be assessed independently to show what dose reduction factor on the DBA dose could credibly be claimed, without repeating and documenting numerous detailed analyses of a range of release scenarios by postulating numerous worst-case events.

An important example of assessing dose estimate assumptions and their uncertainties is the rate of HT conversion to HTO. This is very difficult to estimate and doses are extremely sensitive to the HTO content because of the unique nature of the various tritium DCFs. No justification is provided for the SAR assumption of 25% HTO in the releases, other than to say it is adequately conservative. A worst-case assessment would simply just assume 100% ultimate conversion to HTO, thus simplifying calculations and the need for any conversion rate assumption. If the predictive DBA dose is unacceptable then arguments would be made to derive dose reduction attributes, based on design and operational activities, that are intended to do just that and can be credited to do so.

Referencing and summarizing historical analysis, instead of reproduction in the SAR, would help to reduce excessive volume and much improve readability. The presentation of numerous SAR Appendices makes review difficult if information is not compactly integrated throughout. Content volume could also usefully be much reduced by eliminating information of little safety relevance that could be contained and referenced in lower level technical documents. This should help reduce the frequency of SAR updates. To avoid the necessity of revising large SARs, which requires a significant and dedicated resource commitment, with respect to the most important safety related facility technical limits and administrative controls, the Operating Limits and Conditions (OLCs), it is suggested that these be listed in a stand-alone document of a few pages. The OLCs can than be updated, upon a requirement of the regulator or the facility owner and approved by the regulator, without any need for a licence amendment or a SAR update. The current SRBT licence and LCH indeed currently and satisfactorily defer to the SAR and Section 10 then adequately lists the OLC details. To streamline the approval process for OLC revisions and still maintain regulatory oversight, the SAR only then needs to refer the reader to the current stand-alone OLC document, without any other detail being needed in the SAR.

Omitting other unnecessary operational details (e.g. p. 57 - 59), which have little bearing on the safety case and are in any case documented elsewhere, would also help to streamline the SAR.

3.3 Dose estimate accuracy

A related comment on Figure 12, is that the claim of 3-figure dose accuracy is not realistic, particularly for public doses beyond the facility. Uncertainties in the numerous atmospheric dispersion parameters, the DCF and in other assumptions would be expected to result in no better than an order of magnitude dose estimate, at a distance of 100 m. In-facility staff maximum doses might perhaps be predicted to better than a factor of ten accuracy, but not to the 3-figure accuracy provided in Figure 12. Similarly the ACR-claimed 8-figure accuracy for annually processed tritium and tritium and tritium oxide emissions (5-figure accuracy) seems unduly accurate, although the intervenor is not knowledgeable of the measurement uncertainties.

The 2015 summary comment, made in CMD 15 H5-8 Appendix B "Uncertainties in Dose Estimates" that dose estimates could be very large, still remains valid in my opinion. Addressing the uncertainties of the many contributions to overall dose estimates, or at least recognizing a realistic order of magnitude, would be a longer-term improvement that is recommended be made in the SAR, as well as in the ACR's emissions summary.

3.4 Dose Conversion Factors

The CSA DCF references (Appendices p. 41 of 56) are not easily available for public access. It would therefore would have been useful for the public reviewer for HT / HTO DCF values used to be compared with those of the ICRP, the latter being in the public domain. A comparison of the various literature DCFs would help to identify uncertainties inherent in the values as an important contributor limiting predicted dose accuracy, as a result of tritium's unique and unusual dose properties. Organically Bound Tritium (OBT) doses and associated DCF are not mentioned in the SAR. Numerous mentions of the importance of OBT had been made by intervenors during the 2015 CNSC licence hearings. Lack of mention of OBT seems out of balance with the exhaustive detail and high accuracy provided for a variety of breathing rates (p.42 of 56). Some discussion of the relevance, or otherwise, of OBT in a DBA, would be helpful in a SAR update of dose predictions, especially as the OBT DCF exceeds that of HTO.

Intervenor Conflict Statement

The intervenor has no direct financial interest in SRBT, nor any indirect-financial interests (nearby property ownership, family, personal or professional relationship), nor in any of SRBT's technology. I was the sole author of the 1994 SAR for the Tritium Laboratory, AECL, Chalk River Laboratories, of some technical relevance to the SRBT SAR. The intervenor has a combination of many years of reactor facility operating experience, associated nuclear safety analysis and licensing activities, international reactor facility safety reviews and also employment with the IAEA, Vienna, in the nuclear safety field. The author has consulted on chemical plant process safety and is currently an advisory member of a nuclear safety committee with a national nuclear regulatory body in Europe. As part of preparation for this submission, SRBT accepted my request for a facility tour, which was much appreciated, particularly being cognisant of the inconvenience during COVID.