Feasibility Study on Risk-Informed Reactor Containment Vessels Test Interval Extension in Japan

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Abstract: Nuclear licensees in Japan have announced their decisions of introducing the risk-informed-decision-making (RIDM) process into management processes at nuclear Power Plants with a framework to appropriately assess their initiatives, effectively reduce risks and improve safety, which was stated in "Strategic and action plans for the implementation of risk information utilization at nuclear power stations," in February 2018(latest updated in 2023[1]). In parallel with licensee's efforts of introducing and practicing RIDM process in their plant management, Nuclear Risk Research Center (NRRC) of the Central Research Institute of Electric Power Industry (CRIEPI) works research and development (R&D) related to utilization and improvement of infrastructures for RIDM, ongoing R&D and application of results, and expansion of scope of RIDM process application in order to support the achievement of nuclear safety by the enhancement of PRA.
This paper provides information about precedent US practices of Containment Vessel Leak Rate Test (CVLRT) interval extension and results of feasibility study for Type A (Integrated LRT) test interval extension by risk impact assessment with collected performance data of PWR and BWR plants in Japan. This study aims to introduce US initiative to gain risk benefits by decreasing occupational radioactive exposure and load work by Type A test interval extension up to once test in 15 years. Risk impact assessment results small increase in risk that there are confirmed feasibility.

Keywords: Containment Leakage Rate test program, RIDM, Probabilistic Risk Assessment, Type A test frequency (interval) extension, Risk Management

1. INTRODUCTION

Working on the expansion of RIDM process/ application scope as stated in the licensee's action plans, NRRC promotes activities for leading to compatibility between safety maintenance/improvement and the improvement of a nuclear power plant capacity factor through the risk assessment in operation/maintenance in a plant and conducting resources operation effectively and efficiently based on quantitative risk information.
After the Fukushima accident, NRRC confirmed that some of licensees resuming plant power operation have issues in their predicament situation of the congestion maintenance works that their work is limited only during plant shutdown by the regulation in Japan for many equipment, including additional severe accident measures. To resolve this issue, NRRC set test a interval extension for CVLRT as one of the themes in the RIDM application study.

CVLRT consists of three test types, integrated LRT(as Type A test), local LRT for penetration (Type B test) and isolation valves (Type C test). Focusing on Type A test, performing Type A test is the critical pathway that requires a few days during a periodic maintenance schedule and also needs high manpower cost for preparation and data measurement with shutting out other works inside CV.
In US, 10CFR Part 50 appendix J[2] for CVLRT requirements has option A and B. Option A is the basic traditional regulatory requirement and option B is the performance-based option that licensees can chose one voluntarily. According to these options, the Type A test interval by option A is three tests in 10 years. Option B would be able to extend the test interval up to every 15 years, which licensee can set voluntarily. This initiative is confirmed effective for the reduction of occupational exposure by reducing the number of test times, optimizing plant management resources by reducing the burden of tests and test cost reduction. Nowadays many overseas nuclear plants employ this initiative.
This paper shows the examinations of technical issue and feasibility by introducing Type A test interval extension to Japanese plants with reference to the US practice.
Since the regulatory system in Japan does not have a performance-based option, a regulatory guide that can be used for RIDM process. In this study, risk assessment is performed mainly by referring to use the US regulatory guidelines and related documents.

2. OUTLINE OF CVLRT

2.1. Outline of CV, LRT and system in Japan

Containment vessel (CV) is one of the engineered safety features, which is installed for the purpose of mitigating the diffusion of radioactive material from the reactor core to the environment when the nuclear accident occurrence to ensure public safety around the reactor site.
CV is designed and constructed regarding regulatory requirements such withstanding maximum design pressure, temperature and allowable leakage rate when Loss of Coolant Accident (LOCA) and severe accident occurred.CV boundary is composed of main body and penetration nozzles, pipes and bellows, and isolation valves.

CVLRT is to performed during plant shutdown for refueling that the allowable leak rate values as specified in the plant technical specifications are required not to be exceeded for the assumed events of CV integrity design with isolation function. Industrial test guidance with test methodology, test interval, and other requirements has been published as "Implementation guidelines for Containment Vessel Leak Rate Test (JEAC4203)[3] and interpretation for regulatory technical standards[4] endorses to use JEAC4203 for nuclear power plants in Japan. CVLRT is that CV penetrations or the sealed CV boundary components (airlock and equipment hatch) and the airtightness of a CV isolation valve are tested by locally pressurizing with air or gas, applying greater pressure than that set for the design pressure test for individual equipment, each group (local leak rate test (as Type B for penetrations, air locks and Type C for isolation valves) or whole part of CV boundary(as integrated leak rate test (Type A test)).The images of each test and the test interval requirements are shown in Fig. 1 and Table 1, respectively.

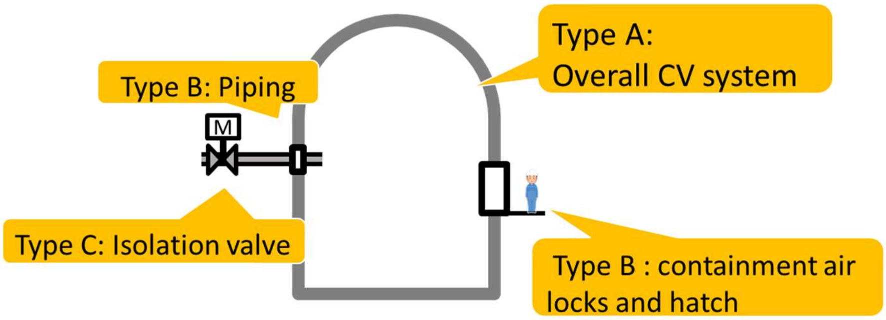

Fig. 1 Schematic chart of CVLRT (Type A, B and C tests)

Table 1 Outline of CVLRT test intervals (Type A, B and C tests) in Japan

Test types	Test intervals
Integrated leak rate test (Type A test)	-Every plant shutdown or every 3 plant shutdowns
Local leak rate test for penetration (Type B test)	-Every plant shutdown or two tests in 3 plant shutdowns - Airlock: every 6 months
Local leak rate test for isolation valve (Type C test)	-Every plant shutdown or two tests in 3 plant shutdowns

* Copy from JEAC4203

Test-related work for CVLRT is carried out during shutdown for refueling. Since Type A test involves long-time work by pressurization and measurement, it corresponds to the condition where other plant inspection works cannot be carried out inside CV (this means critical pathway in inspection schedule) during the test. This critical pathway may require a few days to about one week depending on the period of a series of work by the test procedure preparation, working team establishment, and field work for CV isolation, data measurement, and evaluation test results (Fig.2).

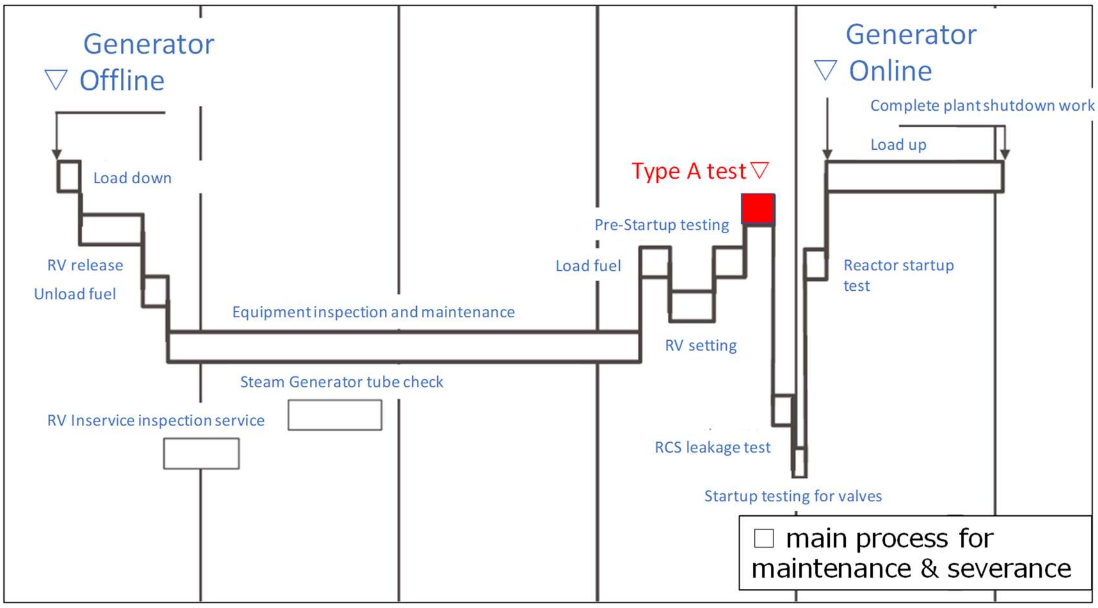

Fig. 2 Image of typical maintenance and inspection work schedule during plant shutdown including Type A test

2.2. Outline of CVLRT in US

In the US case, 10CFR Part 50 appendix J is the basic CVLRT requirements and shows endorsement to use the American National Standards Institute, American Nuclear Society (ANSI/ANS 56.8)[5]. Licensees can choose from two options, A and B.

Option A is defined as a traditional regulatory test interval. Test interval for Type A test by option A is required to perform the test at theree tests per 10 years. Option B is a performance-based option for a test interval. Type A test interval by option B can extend up to every 15 years based on the CV historical performance at a test interval from option A (details are in Table 2) .

Specific RIDM application guidelines are also provided by USNRC as Regulatory Guide1.163[6], by the industry guide of the US Nuclear Energy Institute (NEI) providing test requirements for option B as NEI94-01[7] and risk assessment guideline by the US Electric Power Research Institute (EPRI) as EPRI report 1018243[8].

Specific test requirements on CVLRTs in Japan are based on the technical standards regulations, their interpretation, and JEAC4203. These requirements do not include performance-based regulation items similar to those of in US. A comparison of the CVLRT system in the US and Japan is shown in Fig. 3.

JEAC4203 regulates test intervals and is endorsed by the regulatory body, so that an amended JEAC4203 with the additional new performance-based option is necessary to apply this initiative.

Table 2 CVLRT intervals in US

Test types	Test intervals (Option A)	Test intervals by Performance base (Option B)
CV integrated leak rate test (Type A test)	3 tests per 10 years	Once test in every 15 years at the maximum (need to perform test by total pressure and additional visual inspection at 3 times per extended year)
CV local leak rate test (Type B test)	Except for airlock: Once test in every 2 years Airlock: Once test in every 6 months	Except for airlock: Once test in every 120 months at the maximum Airlock: Once test in every 30 months at the maximum
CV isolation valve local leak rate test (Type C test)	Once test in every 2 years	Once test in every 75 months at the maximum

* Copy from 10 CFR Part 50 Appendix J and relevant information

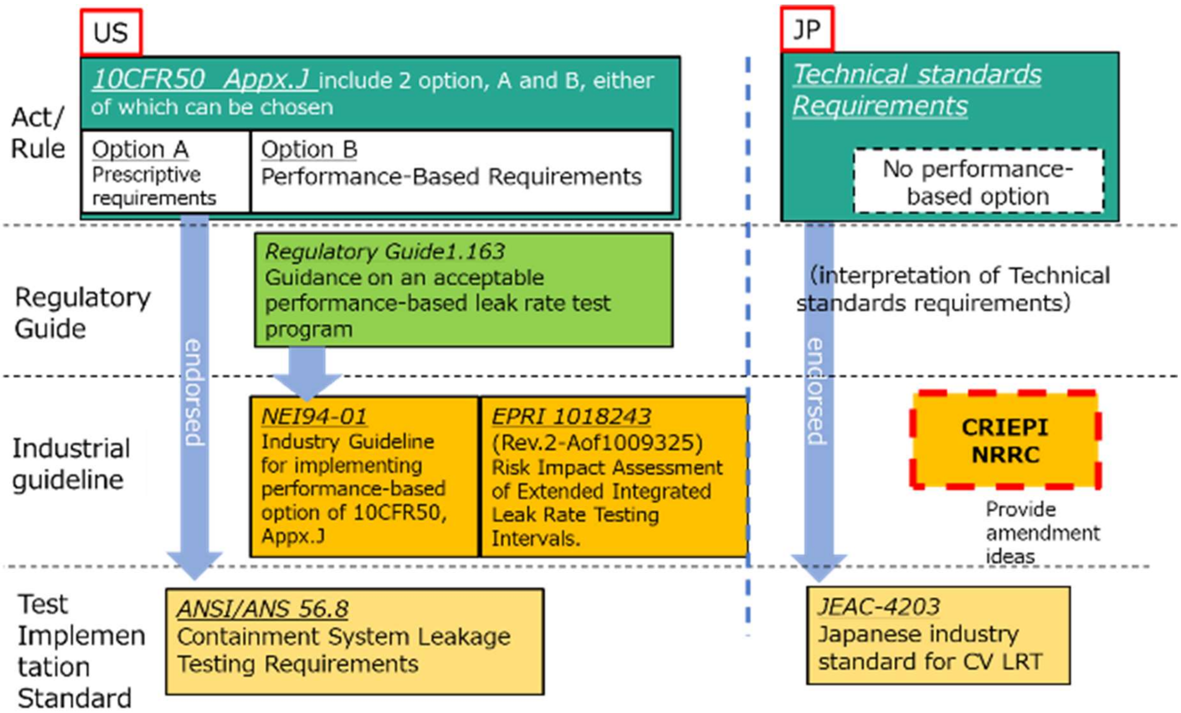

Fig.3 Comparison of relevant CVLRT regulation system in the US and JP

3. RISK IMPACT ASSESSMENT

3.1. Prerequisite conditions

a. Risk metric

EPRI 1018243 report provides methodology steps for the risk impact assessment when introducing Type A test interval extension by assessing large early release frequency (LERF), conditional containment failure probability (CCFP) and population dose. The impact of CCFP at less than 1.5% and population dose were used to be examined for a confirmation of consistency with the USNRC`s Safety Goals before Regulatory Guide 1.174 was published in US risk-informed regulation history. Aligning to current Regulatory Guide 1.174 of USNRC[9] and Japanese risk metrics, LERF should be altered to use Containment Failure Frequency (CFF) in Japan case because CFF is one of the risk metrics evaluated in the process of Japanese Reactor Oversight Process (ROP) that using CFF in RIDM process is able to keep consistency in the risk assessment.

b. Assessment methodology and accident class

EPRI provides the methodology to employ a simplified risk model for this study. It shows that complex containment event tree is not necessary to evaluate the impact of containment isolation system failures and the classification was developed to distinguish between accident sequences affected by the containment isolation system and severe accident phenomena (Table 3). Each LERF frequency is simply calculated by core damage frequency (CDF) times the leakage probability in each accident class, which is defined in Table 3. The change in the leakage probability is detectable only by the Type A test (at class 3a (Small pre-existing leak in containment) and class 3b (Large pre-existing leak in containment)) for the new surveillance intervals of interest. For LERF assessment, class 3b is used in US. More comprehensive evaluations are performed based on the relation between the accident class 3a , class 3b and CV isolation function loss mode (β mode) of Level 2 PRA standards of AESJ[10].

Table 3. EPRI accident class and comparison CV isolation function loss mode with Level 2 PRA standards

EPRI accident classification				Comparison with L2 PRA standard categorization
Class		Frequency	Leakage	
Class 1	Containment intact	F_{Class1}= CDF_{intact}-$F_{Class3a}$-$F_{Class3b}$	La	Not applicable (in the category of design leak)
Class 2	Large containment isolation failures	Value from plant PRA F_{Class2}=$P_{large\ CI}$*CDF_{total}	Value from plant PRA	- Classes 2 to 6 are equivalent to failure in isolating CV (β mode) - Classes 3a and 3b have sensitivity only for Type A test. Also, class 3b is equivalent to LERF -Not applicable because classes 2, 4, 5, and 6 can be detected by Type B and C tests.
Class 3a	Small pre-existing leak in containment	$F_{Class3a}$=$P_{Class3a}$*CDF	10La	
Class 3b	Large pre-existing leak in containment	$F_{Class3b}$=$P_{Class3b}$*CDF	100La	
Class 4	Small isolation failure – failure to seal – (Type B test)	N/A	N/A	
Class 5	Small isolation failure – failure to seal - (Type C test)	N/A	N/A	
Class 6	Containment isolation failures (dependent failures personnel errors)	N/A	N/A	
Class 7	Severe accident phenomena-induced failures (early and late containment failures)	Value from plant PRA F_{Class7}=CDF_{CFL}+CDF_{CFE}	Value from plant PRA	Not applicable (Not β mode)
Class 8	Containment bypass (SGTR, MSIV leakage, and ISLOCA)	Value from plant PRA F_{Class8}=CDF_{ISLOCA} +$CDF_{unisolated\ SGTR}$	Value from plant PRA	Not applicable (Not β mode)

CDF_{intact} = the core damage frequency for intact containment sequences from the plant-specific PRAs
$P_{large\ CI}$ = random containment large isolation failure probability (i.e. large valves)
CDF_{Total} = total plant-specific core damage frequency
$P_{Class\ 3a}$ = the probability of small (10 La) pre-existing containment leakage
$P_{Class\ 3b}$ = the probability of large (100 La) pre-existing containment leakage
CDF_{CFE} = the core damage frequency resulting from accident sequences that lead to early containment failure
CDF_{CFL} = the core damage frequency resulting from accident sequences that lead to late containment failure
La: (percent/24 hours) means the maximum allowable leakage rate at pressure Pa as specified for preoperational tests in the technical specifications.

* Developed based on EPRI document, Level 2 PRA standards of AESJ

3.2. Confirmation of the details of collected test data

a. Data collection

Type A test data of BWR plants and PWR plants in Japan were collected and summarized as follows and Fig.4. These collected plants have been applied or are going to be applied for installation /modification licensing applications based on the new regulatory standards after the Fukushima accident.

- Number of collected plants: 34 in total (PWR:16, BWR:18)
- Number of collected Type A tests performed: 376 times (PWR: 174 times, BWR: 202 times)
- Number of leakage events found: 0

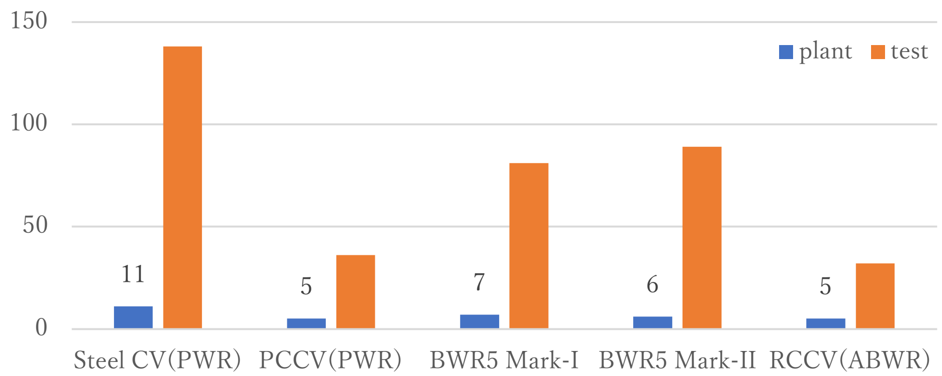

Fig.4 The number of plants and Type A tests in each CV types

According to the collected test data, the averaged test interval was found to be about 38 months for PWR plants and about 18 months for BWR plants. Test data includes a wide range of cases that were actually conducted at longer intervals, which are related to the plant's operation, unplanned shutdowns due to the troubles, and long term shutdown status after the Fukushima accident (Fig.5).
Prior to 1996, all BWR and PWR plants were subjected to perform Type A test at every plant shutdown for refueling.
Since JEAC4203 was amended in 1996 with introducing a new optional Type A test interval at every three plant shutdowns for refueling, it has been employed to perform in all PWR plants. In addition, the data in 1996 to 2021, including the shutdown period of the Fukushima accident, have longer Type A test interval than the averaged test intervals in accordance with JEAC4203 requirements. On the other hand, BWRs have continued to perform Type A test in every shutdown.

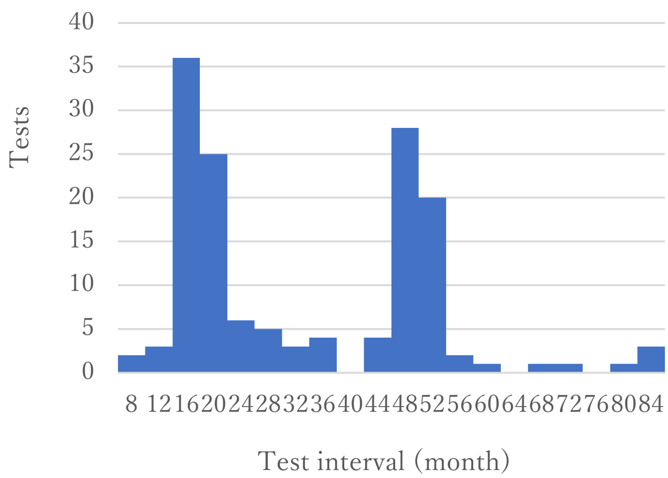

Fig.5 Breakdown of Type A test intervals in PWR dada

No leakage events were found. Some PWR plants have been restarted their power operation recently, and the test results satisfied test criteria. Therefore, it can be confirmed that good performance has been maintained regardless of the actual shutdown period.

3.3. Probability estimation

a. Data screening for the leakage event
In the EPRI 1018243 report, the leakage probability is estimated by using of the actual performances of Type A test results performed by US licensees based on the number of large leak events and is detectable only by Type A test. A leak event is judged detectable by Type B, Type C tests or alternative means caused by operation / maintenance works and is classified as not applicable, as an event that is not affected by the extension of Type A test. Also, in terms of the leakage amount, EPRI conservatively classifies leakage rates as considerable events at from the design leakage rate to approximately 10 La for class 3a and 100 La for class 3b. In assuming the CV performance in Japan, there is no such a classification and requirement for the allowable leakage rate corresponding to classes 3a and 3b. Therefore, in order to set specific values and perform the risk assessment, a test data for screening analysis is required to confirm whether the leakage rate is regarded as the loss of CV isolation function with excess screening value. For cases where the value is exceeded, analysis is continued to perform confirmation whether it can be detected only by Type A test.
In this study, the screening analysis value is referred to using 1.0 La related to the licensing base on CV design when values exceeding these criteria are regarded as a large leakage event in this study.

b. Performing probability estimation
In EPRI report, each LERF frequency is simply calculated by CDF times the leakage probability in each accident class. Similarly, the equation is performed as follows.

$$CFF=(\text{the leakage probability})*CDF_{Baseline}=CCFP_{Baseline}*CDF_{Baseline}$$

Since this method performs CFF assessment regardless of pre-existing containment leakage, the assessment result is conservative. In this case, conservativeness means including a frequency of the loss of CV function due to factors of CV bypass, early and late CV damage modes of early large-scale release other than the pre-existing containment leakage. When the conservativeness is excluded, the calculation is made as follows.

$$CFF = (\text{the leakage probability}) * CDF_{Baseline} - CFF_{\text{without pre-existing containment leakage}}$$

Since this CFF assessment was made regardless early or large-scale release, it corresponds to consider classes 3a and 3b. ΔCFF is found as the changes of risk.

It is confirmed that there is no difference in the leakage probability on the basis of a specific test interval because of zero leakage event. For estimation of a baseline CCFP, all the collected data of Type A test are available to use without test failures, and then Jeffreys non-informative prior distribution model is available to use. The leakage-causing events, which can only be confirmed by Type A test, are in the early stage and their effects will accumulate over the test interval period. To account this test interval period, a time-based standby failure rate model is applied to use. This can conservatively assess the impact of upper bound baseline CCFP. Then baseline CCFP is calculated 1.33E-3.The leakage probabilities in Table 4 are estimated based on the extended from a current Type A test intervals of case1 and case2 to every 15 years according to US practices.

Table 4. Estimated leakage probability

Baseline CCFP	Baseline test interval	The leakage probability per year	Extended to 15 years case	ΔCCFP
1.33E-3	Case 1 (from 4year-test interval)*1	6.63E-4	4.97E-3	3.65E-3
	Case 2 (from about 1.3year-test interval)*2	1.99E-3	1.49E-2	1.36E-2

*Case 1: (PWR case) Considering a current Type A test interval at every 3 shutdowns (13 months operation and 3 months shutdown) for refueling according to JEAC4203 requirements (48 months in total).
3 months means the actual average number of plant shutdown days for refueling among ‘16 to ’20. in resumed power operation plants.

*Case 2: (BWR case) Test interval at every shutdown (13 months operation and 3 months shutdown) for refueling(16 months in total).

3.4.Results of risk impact assessment

Risk assessment in this study uses CDF value at 3.0E-6, which is the representative numerical value from initial Safety Analysis Report (SAR) data, which are issued only operating PWR plants (some of PWR and BWR plants under regulatory review have not issued SAR). The results are confirmed as less than 1E-7.

Table 5. Risk impact assessment results

Risk impact assessment	MAXΔCFF (15-year/time extension case, at power)
Case1 (from 4year-test interval)	1.09E-8
Case 2 (from about 1.3year-test interval)	4.08E-8

4. CONCLUSION

NRRC promotes licensee’s actions on introducing and practicing RIDM processes into their plants. Type A test interval extension is set as one of the RIDM application study theme that would be contribute to resources and workloads optimization.

ΔCFF was assessed as the risk impact assessment related to Type A test interval extension from current test interval cases to every 15 years by use of collected Type A test data.

As a result of the risk impact assessment, it is confirmed that there is a small (less than 1E-7) in risk that this risk-informed application is feasible for domestic plants. However, it is essential to perform individual plant evaluations according to the plant`s characters.

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