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COMPATIBILITY REVIEW OF JAPANESE LARGE COMPONENTS TRANSPORT WITH APPENDIX VII OF TS-G-1.1

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ABSTRACT

The need for large component transport has arisen due to the retirement and dismantlement of some facilities, as well as component degradation requiring replacement to provide for continued operation at other facilities. In some European countries and the US, feed-water heaters, which are classified as SCO-I, have been transported unpackaged as type IP-1, and steam generators, which are classified as SCO-II, have been transported unpackaged under "Special Arrangements". On the basis of these experiences, IAEA has made "Guidance for transport of large components under special arrangements" in Appendix VII of TS-G1.1.

We reviewed the compatibility of typical Japanese large components with the requirements of Appendix VII. First we evaluated inner-surface contamination of a steam generator with the data on the dose rates on the outside surface. In this evaluation, we calculated the activity intake (Q_{INT}), the maximum radiation level on the outside shell of the component, the external radiation level at 3 m and the strength of the tie down system. We also checked the suitable methods for inspection and evaluation

INTRODUCTION

For large radioactive components transport, there is a track record of transporting nearly a hundred of these components in and between the Member States for over 10 years. Based on these experiences, IAEA has been studying the transport regulations revision. In the latest revision, the study results were summarized as an Appendix to advisory material[2], rather than the IAEA Transport Regulation[1].

In Japan, some large radioactive components are replaced and stored in NPPs. For the decommissioning of the future, there is also an option to be transported of unpackaged large radioactive components

We have studied the suitability of large radioactive components in Japan, in parallel with discussion for IAEA transport regulations. This paper reports the results.

COMPATIBILITY REVIEW WITH APPENDIC VII OF TS-G-1.1

The requirements for large components are summarized in Appendix VII of TS-G-1.1. We confirmed the validity of the large component transport in accordance with Appendix VII (VII.19 – VII.37). The results are as follows;

- VII.19: Activity of the component should be due to contamination. We evaluate them with the contaminated amount evaluation method using dose measurements at the outside surface.

The Co-60 contamination on the inner surface of the object is evaluated with the following equations using the calculated and the measured dose rate distribution on the outer surface [3].

$$A_i = \frac{D_k - \sum_{j \neq i} A_j F_{jk}}{F_{ik}} \qquad (1)$$

where

A_i is the activity density of Co-60 (Bq/cm^2) at a divided contamination area *i*,

D_k is the measured gamma dose rate at detector position *k*,

F_{ik} is the calculated dose rate at position k, from unit Co-60 activity density in area *i*.

The dose rate on the surface of the steam generator is 64µSv/h. Using these equations, the contamination of tube (straight section) is calculated to be 1.6 x 10^4Bq/cm^2. It is evaluated to be 2.2x10^4Bq/cm^2 in consideration of the error of measurement (15%) and the nuclide composition error (20%). Since the tube (bent section) is assumed to be twice this level and a water box to be 1.5 times based on an EPRI report, the tube (bent section) and the water box were estimated as 4.4x10^4Bq/cm^2 and 3.3x10^4Bq/cm^2 respectively.

The steam generator, which was measured at this time, had been taken out and kept without system decontamination. On the other hand, the steam generator at decommissioning will be disassembled after system decontamination. The decontamination factor of system decontamination was assumed by referring to the track record in Germany. Since a steam generator starts demolition work after a safe storage period (5 to 10 years), it is assumed that "A steam generator will be taken out from an institution about seven years after furnace stop." The contamination of tube (straight section) is evaluated to be 4.26 x 10^2Bq/cm^2 by referring to the track record in Germany [4]. Since the tube (bent section) is assumed to be twice this level and a water box to be 1.5 times based on an EPRI report [5], the tube (bent section) and the water box were estimated as 8.52x10^2Bq/cm^2 and 6.39x10^2Bq/cm^2 respectively (See Table-1).

Table-1 Contamination density at demolition with system decontamination

		density（Bq/cm^2）
Contamination density on the tube (straight section)	At measurement	2.20x10^4
	At Reactor shutdown	1.49×10^5
	After decontamination	1.07×10^3
	At Transport (7 years after Reactor shutdown)	4.26×10^2
Evaluation	Tube(straight section)	4.26×10^2
	Tube(bent section)	8.52×10^2
	Water Box	6.39×10^2

$Q_{IV} = C \cdot A \times 10^4 = 4.13\text{x}10^{-1}A_2$

where

C is a level of surface contamination (Bq/cm^2)

A is a surface area of an object (m^2)

Table-2 Evaluation of contamination on inner surface of Steam Generator

nuclide	Tube(straight section) Area=$3.84\text{x}10^{+03}m^2$		Tube(bent section) Area= $9.60\text{x}10^{+03}m^2$		Water Box Area$7.95\text{x}10^{+03}m^2$	
	Density (Bq/cm^2)	Ratio of A_2	Density (Bq/cm^2)	Ratio of A_2	Density (Bq/cm^2)	Ratio of A_2
C-14	$2.35\text{x}10^{+02}$	$3.01\text{ x}10^{-03}$	$4.70\text{ x}10^{+02}$	$1.50\text{ x}10^{-02}$	$3.53\text{ x}10^{+02}$	$9.36\text{ x}10^{-03}$
Fe-55	$5.47\text{ x}10^{+02}$	$5.24\text{ x}10^{-04}$	$1.09\text{ x}10^{+03}$	$2.62\text{ x}10^{-03}$	$8.21\text{ x}10^{+02}$	$1.63\text{ x}10^{-03}$
Co-60	$4.26\text{ x}10^{+02}$	$4.09\text{ x}10^{-02}$	$8.52\text{ x}10^{+02}$	$2.04\text{ x}10^{-01}$	$6.39\text{ x}10^{+02}$	$1.27\text{ x}10^{-01}$
Ni-59	$6.33\text{ x}10^{+00}$		$1.27\text{ x}10^{+01}$		$9.49\text{ x}10^{+00}$	
Ni-63	$7.24\text{ x}10^{+02}$	$9.27\text{ x}10^{-04}$	$1.45\text{ x}10^{+03}$	$4.62\text{ x}10^{-03}$	$1.09\text{ x}10^{+03}$	$2.88\text{ x}10^{-03}$
total		$4.54\text{ x}10^{-02}$		$2.27\text{ x}10^{-01}$		$1.41\text{ x}10^{-01}$

- VII.20 : Intake at accident will be about $4.13\times10^{-7}A_2$. It is derived from the following equation and it is less than $10^{-6}A_2$:

  $Q_{INT,} = Q_{IV}\ x\ F_{SCRAP}\ x\ F_{REL}\ x\ F_{RSUS}\ x\ F_{INT}$$=4.13\times10^{-7}\ A_2$.

Table-3 Evaluation parameter for an activity intake for a person in an accident

parameter	Explanation	value	remark
Q_{IV}	Inventory in a package or an object (Bq)	$4.13\text{x}10^{-1}A_2$	Contamination on inner surface
F_{SCRAP}	The fraction of surface area that scraped in an accident	1	VII.11
F_{RSUS}	The fraction of the released activity which is in a form of respirable aerosol	1	VII.11
F_{REL}	The fraction of the activity which is released from package or the object in an accident	$1\text{x}10^{-2}$	VII.11
F_{INT}	The fraction of respirable released activity intake for a person in the vicinity of the accident	$1\text{x}10^{-4}$	VII.11

- VII.21: According to the nuclide analysis, the presence of fissile nuclide is negligible.
- VII.22: There are no plans with unnecessary, extraneous material.
- VII.23: Dry process should be added in the demolition work.
- VII.24: Radiation level on the outside surface will be less than 64 μSv/h(< 2 mSv/h).

  The dose rate distribution on the outer surface is evaluated with following equations using the calculated and the measured Co-60 contamination on the inner surface of the object. A radiation transport calculation which models the object apparatus by the Monte Carlo method is performed, and the dose rate distribution on the outer surface is calculated based on the equation (1).

The tube is treated by a model in which the internal air and the tube are uniform, and the radiation source is uniformly in this domain. The next assumption is used about source distribution.

- The contamination distribution of the water box is homogeneous.
- The contamination distribution is uniform in the straight section and the bent section of the tube.

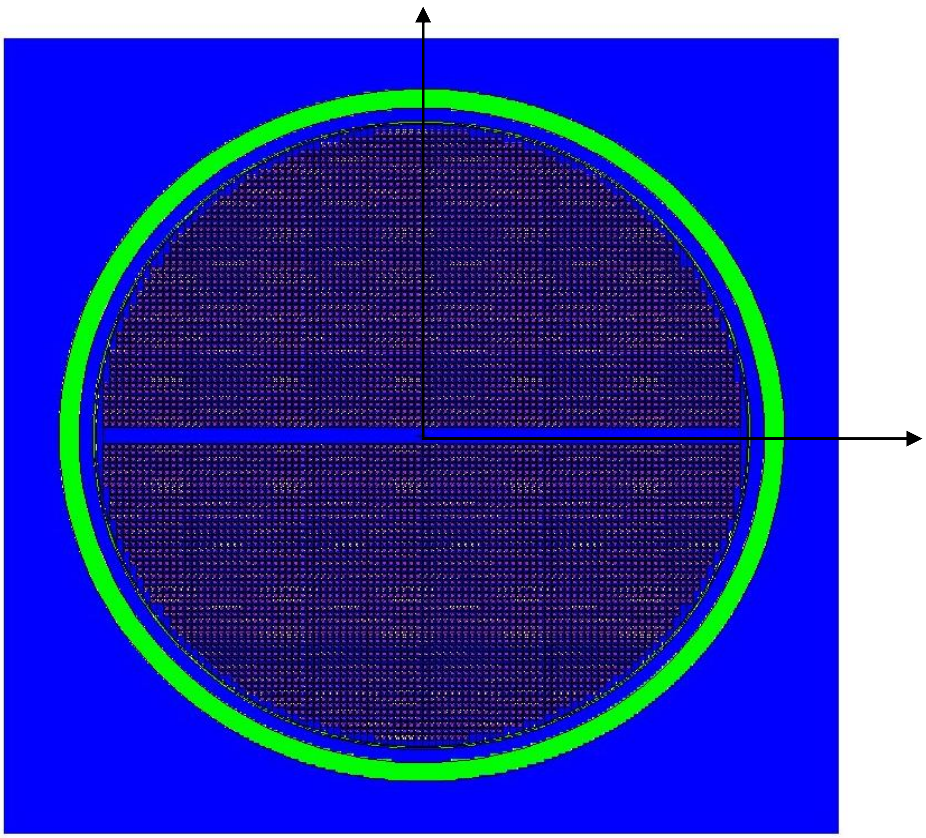

Figure-1 Set of tubes in Steam Generator

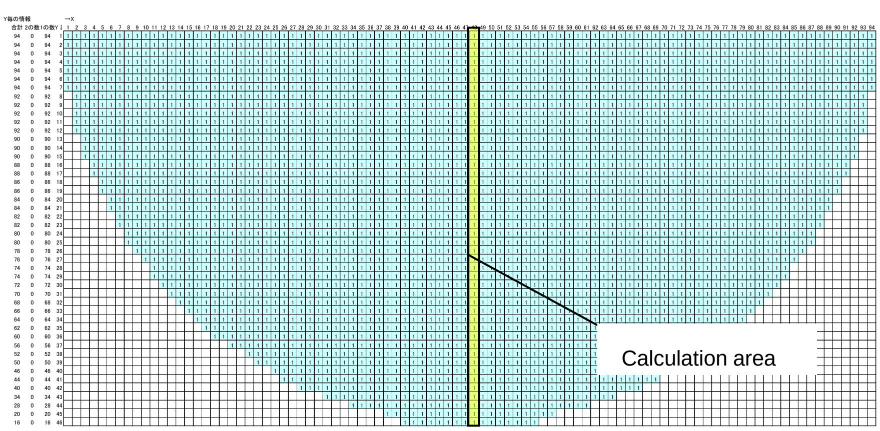

Figure-2 Calculation model for Steam Generator

Table-4 shows dose rate of normal condition of transport.

Table-4 Dose rate of normal condition of transport (Unit: μSv/hr)

Evaluating point	Head		Center of Side	Bottom	
	Axis	Radial		Axis	Radial
surface	—	16.0	64	2.5	—
1m from surface	3.1	14.1	28.1	14.3	1.1

- VII.25: For the steam generator, the external radiation level is calculated to be 1.37 mSv/h for the following conditions. It is less than 10 mSv/h.

  [Conditions]: See Figure-3
  - The radiation source domain is uniform.
  - The intensity of the radiation source is $8x10^5$Bq/cm^2 as the SCO-II maximum.
  - The outer shell and the internal pipe are disregarded.

  [Result]
  - The external radiation level at 3m is 1.37mSv/h. It is less than 10mSv/h

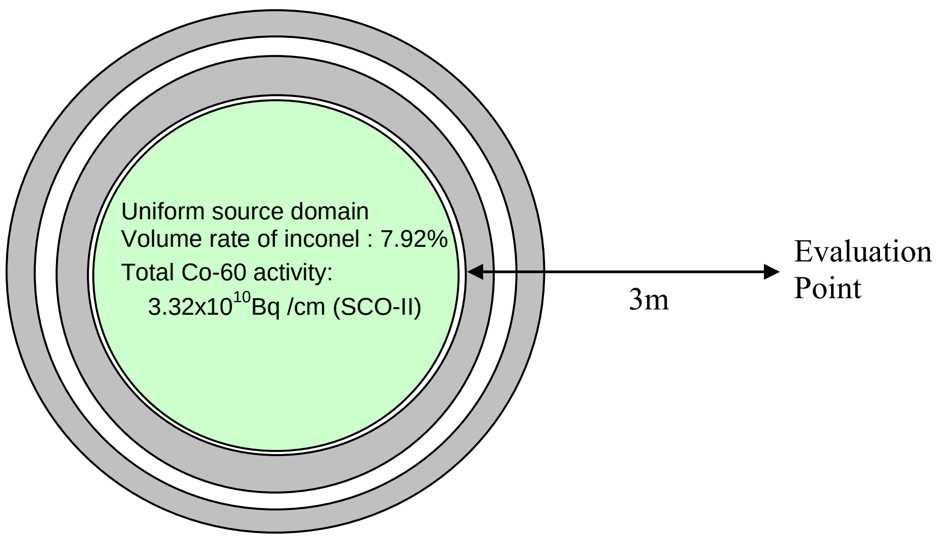

Figure-3 Cross-section model of Steam Generator

- VII.26: The target steam generator should be capable of withstanding the effects of any acceleration, vibration or vibration resonance which may arise under routine conditions of transport.
  [Conditions and result]
  The target steam generator is about 300 tons in gross weight, about 21 m in length, and about 4.5 m (upper) and 3.4m (lower) in diameter. Table-5 shows the conditions for analysis. Table-6 shows stress evaluation criteria for structural analysis. The results are as follows;
  (a) Structural analysis of suspending device and the tie down system
  It is verified that permanent modification did not arise in the suspending device and lashing jig with any acceleration under routine conditions of transport.
  (b) Structural analysis for free drop test
  The deformation and acceleration are evaluated to be less than 20% of stress evaluation

criteria.

Table-5 Conditions for Analysis

Parameter	Conditions
—Acceleration at operation	3g
—Acceleration at transport	
・warp direction	2g
・vertical direction	2g
・lateral direction	2g
—Maximum internal pressure	Atmospheric
—Maximum external pressure	Atmospheric
—Body temperature	25 °C

Table-6 Stress evaluation criteria for structural analysis

<table>
<tr><th>Condition</th><th>Item</th><th>Location</th><th>Pm(PL)</th><th>PL＋Pb</th><th>PL＋Pb＋Q</th><th>PL＋Pb＋Q＋F</th></tr>
<tr><td rowspan="4">Routine condition of transport</td><td>Suspended load</td><td>Trunnion</td><td>＜Sy</td><td>＜Sy</td><td>—</td><td>—</td></tr>
<tr><td>Lashing load</td><td>Lashing jig</td><td>＜Sy</td><td>＜Sy</td><td>—</td><td>—</td></tr>
<tr><td>Pressure</td><td>Body</td><td colspan="4">Capable of withstanding the variation of the ambient pressure.</td></tr>
<tr><td>Vibration</td><td>Package</td><td colspan="4">Capable of withstanding the vibration during transport</td></tr>
<tr><td rowspan="10">Normal condition Of transport</td><td rowspan="3">Thermal test</td><td>Body</td><td>＜Sm</td><td>＜1.5Sm</td><td>＜3Sm</td><td rowspan="3">—</td></tr>
<tr><td>Lid</td><td rowspan="2">＜Sy/1.5</td><td rowspan="2">＜Sy</td><td rowspan="2">＜Sy</td></tr>
<tr><td>Bolt</td></tr>
<tr><td>Water spray test</td><td>Package</td><td colspan="4">Capable of withstanding spraying of water</td></tr>
<tr><td rowspan="3">Free drop test</td><td>Body</td><td>＜Sm</td><td>＜1.5Sm</td><td>＜3Sm</td><td>—</td></tr>
<tr><td>Lid</td><td rowspan="2">＜Sy/1.5</td><td rowspan="2">＜Sy</td><td rowspan="2">＜Sy</td><td>—</td></tr>
<tr><td>Bolt</td><td>—</td></tr>
<tr><td rowspan="2">Compressive load</td><td>Body</td><td>＜Sm</td><td>＜1.5Sm</td><td>＜3Sm</td><td rowspan="2">—</td></tr>
<tr><td>Lid</td><td>＜Sy/1.5</td><td>＜Sy</td><td>＜Sy</td></tr>
<tr><td>Penetration test</td><td>Body</td><td colspan="4">Penetration strength</td></tr>
</table>

- VII.27: It is evaluated at VII.26, and it should meet the IP-2 requirements.
- VII.28: Contamination of the surface outside the components can become low enough by decontamination.
- VII.29: The component should be consigned as exclusive use.
- VII.30: Vehicles transportation and marine transportation are planned.
- VII.31: Transport index would be 28 (=0.0281x100x10) since the external radiation level at 1m is 28.1μSv/h.
- VII.32: The regulations should be followed.

- VII.33: Radiation level on the outside surface would be less than 64 μSv/h (< 2 mSv/h) with the evaluation of VII.24.
- VII.34: It is evaluated at VII.26.
- VII.35: This is carried out according to the management system of the NPP.
- VII.36: The transport orientation restriction based on the transport plan would be considered.
- VII.37: It corresponds to the procedure of "special arrangement".

CONCLUSIONS

In this study, we have reviewed the compatibility of some Japanese large components with the requirements of Appendix VII. This is one example of evaluation for the steam generator which will be generated at the time of decommissioning, and we have established the outline of the compatibility review of Japanese large component transport with Appendix VII of TS-G-1.1.

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