Technical Requirements for Dry Storage Demonstration System of CANDU Spent Fuel

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1. Introduction

The purpose of the CANDU Spent Fuel(CSF) dry storage demonstration study is to obtain empirical data to demonstrate heat removal performance, demonstrate radiation shielding performance, prove health of spent nuclear fuel cladding, and prove recoverability of spent nuclear fuel during operation of the facility. This is because the temperature distribution and environment in the dry storage container are factors that have a decisive influence on the soundness of spent nuclear fuel to be stored. In addition, a sufficient review should be made of the deterioration mechanism expressed in the presence of residual moisture, reflecting the effect of vacuum drying, one of the important processes of dry storage. In order to accurately check the change experienced during the storage period, it is necessary to clearly check the condition of the fuel rod before storage, so data on this should be secured.

2. General information

The demonstration system shall be designed to ensure that dry storage facilities safely and efficiently handle, store, shield, cool and store spent nuclear fuel. The demonstration system shall be able to demonstrate all the loads and conditions that may accompany the dry storage facility under normal operating conditions and abnormal conditions. The CSF stored in the demonstration system shall be stored in a safe recovery manner at any time. Demonstrations of means to safely handle, store and recover damaged CSF should also be included.

Item	Requirements
Wear and tear of support structures	The height of the Bearing Pad should not decrease by more than 0.3mm
Deformation of cladding	Ductility limit value should be limited to less than 3%. Internal pressure of fuel rods
Internal pressure of fuel rods	If the internal pressure of the fuel rod is maintained below the coolant pressure, the primary stress of the cladding becomes compressive stress, excluding damage due to tensile stress.
Wear of spacers	Type 7 or less Type 7 : Estimated depth of wear >1/4 of initial pad thickness (0.70mm in height)
Stress and deformation of fuel rod end plugs	Soundness should be maintained under loads such as hydraulic drag, ram load (tool for fuel loading/unloading), coolant pressure, internal pressure of fuel rod, and pellet expansion
SCC at fuel rod end plug-cladding junction	SCC at fuel rod end plug-cladding junction SCC caused by stress concentration should not occur at the fuel rod end plug-cladding junction
Fatigue of cladding	The fatigue evaluation result of the cladding should satisfy the fatigue limit restriction proposed by the O'Donnell-Langer model
Collapse of cladding	Formation of axial wrinkles on the cladding should be within 0.5%.
Oxidation of cladding	Thickness of the oxide layer should be limited to avoid excessive oxidation.
Hydrides in fuel rods	The total hydrogen content inside the fuel rod should be limited to less than 0.8 mg
Gap corrosion between support/spacer and cladding	Damage to the cladding due to gap corrosion at the brazed area between the support (or spacer) and the cladding should not occur.

Table. Technical requirement

3. Basket design

The mechanical construction of the nuclear fuel basket shall ensure that the mass of the other fully loaded baskets is supported without any structural deformation causing handling problems (where a number of other baskets are loaded on the basket and static, shock and seismic loads must be taken into account). It shall be designed to ensure that CSF is normally located in the basket and that decay heat transfer to the surrounding area meets the design requirements specified for the maximum storage temperature of CSF

4. Measurement of CSF appearance

The length of the CSF shall be measurable. For this purpose, It shall be possible to check the bending of the nuclear fuel through remote visual inspection. During the storage period, gas samples shall be analyzed to confirm fission gas leakage into the air inside the storage cask. In addition, remote visual inspection should be able to identify signs of cracking and corrosion on the surface of the cladding [1]. The degree of crud coating on the surface of the nuclear fuel cladding shall be confirmed through remote visual inspection using a video camera. It shall be possible to measure and verify the physical characteristics of CSF reactors through PSE (Pool Side Examination), etc.

5. Cladding inspection items

The degradation mechanism capable of degrading the performance of the cladding shall be sufficiently provided and reviewed, mainly due to temperature, pressure in the rod, stress in the cladding, and environment during dry storage or transportation [2]. In the case of long-term storage of CSF (more than 20 years), it shall be possible to produce empirical data on the deformation of creep after long-term storage of CSF cladding. The condition of the cladding shall be scrutinized through destructive inspection.

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