Methodology to incorporate damaged BWR spent fuel as contents in the NAC-UMS® storage system

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Abstract. Following Taiwan’s nuclear waste management policy, INER launched Taiwan’s first Independent Spent Fuel Storage Installation (ISFSI) project in 2005. The technology of dry storage system adopted is NAC-UMS system. INER made some design modifications of NAC-UMS system in order to accommodate the constraints at Chin-Shan Nuclear Power Plant (CSNPP). The modified UMS was named as INER High Performance System (INER-HPS). The corresponding Safety Analysis Report (SAR) of ISFSI was approved by AEC on November of 2008. Through years of dedication, the first dry run test has been accomplished on November 2012. INER-HPS is capable to cover most of the spent fuels dry storage needs in Taiwan with current NAC-UMS License. The current CoC for NAC-UMS includes undamaged BWR fuel with a burn-up of not greater than 45 GWD/MTU. For the purpose of NAC-UMS to incorporate damaged BWR fuel as approved contents, INER and NAC further enter into cooperation to developing and licensing NAC-UMS to include up to four BWR damaged fuel cans and a new BWR damaged fuel basket. This paper summarized the project progress, results of trial fabrication of damaged fuel can, and the methodology to incorporate damaged BWR spent fuel as contents in the NAC-UMS storage system.

1. Introduction
In 2005, The Independent Spent Fuel Storage Installation (ISFSI) project was launched at Institute of Nuclear Energy Research (INER) following the nuclear waste management policy of ROCAEC. Taiwan Power Company (TPC) has entrusted the INER to establish a dry storage facility, and NAC was selected by the Institute for Nuclear Energy Research (INER) to provide technology for the first application of dry spent fuel storage at nuclear power plants in Taiwan. The prototype of dry storage system is adopted of NAC-UMS system. NAC International designed and licensed the UMS® - Universal Multipurpose Systems – to handle virtually all types of Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) fuel used in commercial nuclear power plants in the United States and around the world. The UMS system is an evolutionary design based on the earlier-licensed NAC-STC basket and transport overpack designs. It has a capacity of 24 PWR or 56 BWR fuel assemblies and provides a single system that embodies the low cost of a storage-only system, the transportability of a dual-purpose canister-based system, and the canister benefits envisioned in the discontinued U.S. Department of Energy (DOE) Multi-Purpose Canister (MPC) program.

The contract between INER and NAC included the licensing of the UMS technology to INER, support for licensing of the BWR UMS-56 with Taiwan’s Atomic Energy Council, application of the UMS-56
for dry storage at Taiwan Power Company’s Chinshan Nuclear Station, and support for joint development and further application of UMS technology. Some designs were modified in order to accommodate the site constraints of CSNPP. NAC also supported INER in the design of a lightweight transfer cask to be used at Chinsan, which is limited by a 98-ton reactor-building crane. The modified UMS system was also named as INER High Performance System (INER-HPS). INER-HPS fulfills most of the needs of the spent fuels dry storage in Taiwan with current license. With further License amendment to extend BWR fuel average bundle burnup up to maximum 60 GWd/MTU as well as damaged fuel storage, the INER-HPS system can cover all the dry spent fuel storage issues in Taiwan, and a quickest means to achieve the goal of decommissioning of nuclear power plant.

2. Project scope
NAC International (NAC) and The Institute of Nuclear Energy Research (INER) agreed to submit a UMS CoC amendment request to the USNRC to incorporate High Burnup BWR Fuel and BWR Damage Fuel as approved storage contents. The joint staff performed design and analytical activities according to an agreed to division of responsibilities. Following completion of design and analysis activities, NAC submitted the licensing package as an amendment to the current USNRC UMS CoC. Both NAC and INER staff provided technical support for the NRC review, including preparing responses to NRC requests for additional information. In this project, NAC carried the preparation of Damaged Fuel Can (DFC) and canister design modification, SAR and Technical Specifications update and CoC amendment submittal. With preparation of calculation and reports, including Criticality, Shielding, Thermal, Structure analysis supporting by INER.

This project is a 2 years period from June 1st 2019 to November 30th 2021. The works have completed the following: preparing of supporting reports, revising of Safety Analysis Report (SAR) and Technical Specifications (TS), Cooperating together with NAC and INER to Amendment of Certificate of Compliance (CoC) No.1015 for obtaining the approved contents of NAC-UMS® Dry Storage System to incorporate High Burnup (HBU) and Damaged BWR Fuel (DF). The request to revise associated bases in the FSAR has been submitted including damaged BWR fuel as authorized contents on December 18, 2019. The NRC notified that the application contains enough information to conduct a detailed technical review and the acceptance review for the application has accomplished on March 17, 2020. On June 25, 2020 the NRC issued an RAI with responses are due by August 25. And they were submitted on August 7. NRC completing the safety review of the application, and on February 23, 2021 the proposed CoC, Technical Specifications, and preliminary SER were placed into ADAMS for preparation of a rulemaking package.

3. Design changes
The NAC-UMS® system is a dry storage system whose principal components are a transportable storage canister (TSC), vertical concrete cask (VCC), and transfer cask. NAC’s Safety Analysis Report (SAR) has demonstrated the ability of the UMS satisfies the requirements of the U.S. Nuclear Regulatory Commission (NRC) for the storage of spent nuclear fuel as prescribed in Title 10 of the Code of Federal Regulations, Part 72 and NUREG-1536. The transportation component of the UMS® is designated the Universal Transportation System, which is addressed in the NAC Safety Analysis Report for the Universal Transport Cask, Docket No. 71-9270.

The UMS primary components consist of the Transportable Storage Canister (TSC), Vertical Concrete Cask (VCC), and a transfer cask (TFR). In long-term storage, the TSC is installed in a VCC, which provides passive radiation shielding and natural convection cooling. The VCC also provides protection during storage for the TSC under adverse environmental conditions.

The TFR is used to move the TSC during storage processing, where the canister is loaded with fuel (figure. 1), welded closed, backfilled with inert gas and then placed inside the VCC. The TFR also used to transfer the canister from the VCC to the Universal Transport Cask for transport.

The fuel basket design is based on the standard Fuel Basket but with the four corner fuel tubes replaced by a BWR Damaged Fuel Can (Figure 2). To accommodate the DFCs, the corner openings in the upper and lower weldments have larger, 6.43-inches square openings. The BWR DFC design is 6.30-inches inside square body and incorporates two neutron absorber sheets as the fuel tubes replaced. The primary
support disk evaluation performed in adopting the existing licensed BWR fuel basket to accommodate damaged fuel cans required addressing a slight increase in corner position loads due to slightly heavier tubes and the inclusion of a bottom plate and lid.

To allow the DFC’s to be placed in the basket, the upper and lower weldments required much larger openings than the FSAR standard design, resulting in the section reduction of the ligaments in the upper and lower weldment corners.

As such, no modifications were required for the support disks or thermal fins. The top and bottom weldment corner position openings as noted above were enlarged to allow the DFC’s to pass thru and was evaluated for the Amendment.

Figure 1. Underwater handling of fuel assembly to storage canister.

Figure 2. Schematic diagram of DFC.

4. Structure, thermal, criticality and shielding evaluation

This section summarizes the evaluation results of structure, thermal, criticality and shielding, which make sure DFC can meet what are required in the regulations

4.1. Basket structural evaluation

The BWR fuel basket support disks and weldments were evaluated for normal, off-normal and the 24-inch drop accident conditions. The BWR fuel basket support disks are also evaluated for the tip-over condition.

Although thermal stresses are also evaluated for normal and off-normal conditions, the maximum basket temperature for the BWR5 DF configuration is bounded by the maximum temperature for the BWR5 configuration (all fuel assemblies are undamaged). Therefore, the thermal stress evaluation for the basket in the FSAR remains bounding.

The support disks have the identical design for both the BWR5 and BWR5 DF configurations. The governing load condition for the support disks is the side impact (in the in-plane direction of the disk). For the off-normal conditions, the analysis in the FSAR uses a maximum weight (per disk) of 1,095 lb, which is greater than that for the support disk for BWR5 DF basket (1,076 lb). For the accident condition of tip-over, the analysis for BWR support disks in the FSAR uses a load of 32.6 kips per disk (30g), which is greater than the load of 32.4 kips per disk (30g) for the support disk for BWR5 DF basket. Therefore, the evaluation of support disks for normal, off-normal and accident conditions in the FSAR are bounding.

The top weldment for the BWR DF configuration is based on the top weldment for BWR configuration with the opening of the four corner slots increased from 5.90 inches × 5.90 inches to 6.428 inches × 6.428 inches. The governing load condition for the top weldment is the 60-g end impact for the 24-inch cask drop accident. The sectional stress intensity at section Stw of the ligament for the top weldment of BWR DF configuration is determined to be 17.0 ksi (=12.1×1.178/0.839), using the ratio of the ligament widths of both configurations. Since the maximum stress intensity of the ligaments for the corner slots of the top weldment for the BWR DF basket (17.0 ksi) is well below the maximum stress intensity (34.1 ksi) of the top weldment for the BWR basket for undamaged fuel as currently presented in the FSAR, further analysis was required.
Similar conditions applied to the bottom weldment, which also received an increased corner slot size, where the width of ligaments adjacent to the slot are reduced and the stresses in the ligaments require evaluation. Based on the stress results from the FSAR analysis, the maximum sectional stress intensity for the ligaments of corner slots is 27.6 ksi. Using the same method as the top weldment evaluation, the sectional stress intensity for the affected sections of corner slot ligaments was determined to be 39.8 ksi. Since the maximum stress intensity of the ligaments for the corner slots of the bottom weldment for the BWR DF basket (39.8 ksi) is well below the maximum stress intensity (51.9 ksi) of the bottom weldment for the BWR basket for undamaged fuel.

### 4.2. Basket thermal evaluation

The evaluation assumed 100% failure of the fuel rods and cladding in the DFCs with 50% compaction of the fuel debris. The decay heat for a single fuel assembly (0.411 kW) is concentrated in the debris region located at the lower part of the DFCs. Total thermal loading of one BWR canister is not changed (23 kW). The three-dimensional BWR canister model is applied for the evaluation for BWR5 DF configuration. The maximum temperatures of the fuel cladding, damaged fuel, support disks, and heat transfer disks for the two analyzed cases in the following Table 1. The results shows that modelling the fuel debris region located at the lower part of the active fuel region is acceptable for the thermal analysis because this location is closer to the center of the basket where the maximum fuel cladding temperature occurs. The maximum temperatures of the fuel cladding, damaged fuel, support disks, and heat transfer disks are bounded by the corresponding temperatures for the design basis BWR fuel and are below the allowable temperature limits. Therefore, the use of the BWR DFC to contain damaged fuel or fuel debris is acceptable under normal conditions of storage and transfer operations and will meet the requirements in 10 CFR 72.236(f).

| Table 1. Summaries of Maximum Temperatures for Cases 1 and 2. |
|---------------------------------------------------------------|
| **Maximum Temperature of component** (°F) | **With 4 BWR DFCs** | **Normal Condition** | **Allowable** |
| Intact Fuel | 637 | 642 | 752 |
| Damaged Fuel | 519 | N/A | 752 |
| Support Disk | 607 | 614 | 700 |
| Heat Transfer Disk | 606 | 612 | 650 |

*added 2°F for margin

The BWR system internal pressures during normal, off-normal and accident conditions associated with the addition of the four BWR DF cans were also evaluated. The maximum operating pressure under the
normal conditions (MNOP) of storage and comparison of fission gas inventory, canister free volume, and released fission gas between the design-basis BWR canister and the BWR Class-5DF canister, there is reasonable assurance that the MNOP (3.96 psig) of the BWR5 DF canister, loaded with four DFCs to store fuel or fuel debris, will be below the design limit of 15.0 psig for normal storage. The pressure bounding correlations of the design-basis BWR canister can be extended to the BWR5 DF canister under off-normal and accident conditions of storage and transfer operations. The maximum pressures for off-normal and accident conditions will be below the design limit of 65 psig.

4.3. Structural evaluation for BWR damaged fuel can
The BWR damaged fuel can is designed to accommodate BWR damaged fuel (DF). The DFC fits within the corner slots of the BWR5 DF basket. The primary function of the BWR DFC is to confine the fuel material within the can to minimize the potential for dispersal of the fuel material into the canister cavity volume.

The DFC walls consist of 0.05-inch thick Type 304 stainless steel sheet (18 gauge) with a total length of 177.57 inches. The DFC weldment has a bottom plate that is 0.63 inches thick. Four holes in the bottom plate, filter-screened with Type 304 stainless steel wire screens (with 250 openings/inch × 250 openings/inch mesh, each supported by backing screens with wire mesh having 16 openings/inch × 16 openings/inch), permit water to be drained from the can during loading operations. Since the bottom surface of the DFC rests on the canister bottom plate, additional slots are machined in the DFC (extending from the holes to the side of the bottom assembly) to allow the water to be drained from the can. At the top of the DFC, the top flange extends beyond each of the four walls to allow the use of a handling tool to lift the can and contents.

The structural evaluation of the BWR DFC determines that it is structurally adequate for all conditions of handling and storage, including accident conditions. In normal operation, the can is in a vertical position. The weight of the DFC contents is transferred through the bottom plate of the can to the canister bottom plate, which is the identical load path for undamaged fuel. The only normal operation loading in the vertical direction is the combined weights of the DFC tube body, top flange, and lid assembly. The lifting of the can with its contents is also in the vertical direction.

Classical hand calculations are used to qualify the stresses in the BWR DFC. Calculated stresses are compared to allowable stresses in accordance with ASME Section III, Subsection NG. Bounding accelerations of 60g (end impact) and 30g (side impact). The DFC is also evaluated for normal handling loads including a 10% dynamic load factor. The DFC lifting structure is designed with a load factor of 6 on material yield stress and 10 on material ultimate stress.

4.4. Criticality evaluation
Criticality evaluations were designed to update the UMS FSAR to modify the payload definition, i.e., fuel assembly type and dimensions, while adding damaged fuel capability. Three dimensional models, as shown in figure 4, of the UMS system (Basket, TSC, VCC, and Transfer Cask) had built for evaluation in MCNP6.2, while previous UMS criticality evaluations were performed in old version KENO and MONK.

First criticality evaluations were performed on the nominal UMS basket to accommodate additional fuel types. Three additional fuel types were added, which include two 10x10 assemblies. The absorber sheets in the basket were evaluated at 75% and 90% absorber credit. Previous evaluations were limited to 75% absorber credit. The nominal basket was also evaluated with preferential loading, i.e. loading assemblies with lower initial enrichment fuels to enable loading higher initial enrichment fuels at certain peripheral positions. The preferential loading was limited to 75% absorber credit.

The damaged fuel basket was evaluated with 52 undamaged fuel assemblies and the four damaged assemblies in DFCs in the corners of the basket. Three variations are applied to evaluate the fuels in the DFCs. The first evaluation is undamaged assemblies loaded into the DFCs. The next variation is the unclad array, which is a fuel rods array without clad and nozzles but with a variable rod pitch, aim to simulate possible configuration of damaged fuel lose its structure support in the DFC. Finally, the mixture case which models a homogenized mixture of fuel and moderator with various volumes of moderator to ensure a bounding H/U ratio, aim to simulate the fuel pellets drop out from cladding and
accumulate at the bottom of DFC. Several studies are also made to examine the influence of moderator condition, i.e., variable moderator density at internal and external of DFC, preferential flooding and partial flooding. The maximum initial enrichment was evaluated for fuel in the damaged fuel basket with 75% and 90% absorber credit. Enrichment limits were set based on evaluation results to remain below the upper subcritical limit (USL). The USL is determined by benchmark MCNP6.2 with 181 criticality experiments. In summary with these evaluation results, loading with four DFCs and damaged fuels will not lead a significant reactivity increase comparing to original payload with 56 undamaged fuels, as shown in Table 2. While some evaluations indicate a reduced enrichment requirement for some of the fuel types, this is caused by the statistical fluctuations in the MCNP.

**Table 2. Maximum Initial Enrichment with four DFCs loaded with damaged fuel.**

| Fuel Type | Maximum Initial Enrichment (\(^{235}\text{U}\) wt%) |
|-----------|-----------------------------|
|           | 75% Neutron Absorber Credit | 90% Neutron Absorber Credit |
| ge08n     | 4.80                        | 5.00                        |
| ge08k     | 4.70                        | 4.90                        |
| ge08i     | 4.60*                       | 4.90                        |
| ex08b     | 4.70                        | 4.90                        |
| ex09c     | 4.50*                       | 4.70                        |
| B9_72A    | 4.40*                       | 4.60*                       |
| B10_91A   | 4.40*                       | 4.60*                       |
| B10_92A   | 4.40                        | 4.60                        |

* The asterisk indicates that the Maximum Initial Enrichment for payloads with four DFC is lower by 0.1 wt% than payload with 56 undamaged fuels.

**4.5. Shielding evaluation**

Shielding evaluations include the additional high burnup (HBU) source evaluation, as well as the damaged fuel basket dose rate evaluations. The dose rate evaluations include the single cask evaluation, site dose calculations and occupational dose calculations. High burnup is any amount over 45 GWd/MTU. The HBU evaluation includes burnup up to 60 GWd/MTU. Assembly source terms evaluations were carried out by TRITON-ARP. Dose rate assessments were carried out by MCNP. Damaged basket dose rate assessments were performed for both the TFR and VCC with the bounding case determined by HBU dose rate evaluations. Damaged fuel is calculated by assuming that the fuel assembly has collapsed in the DFC. This bounds loading debris into DFC up to an assembly worth. Packing fractions up to a maximum of 0.75 were considered. The packing fractions and associate fuel debris heights are listed in Table 3. The hardware was assumed to stay in place. The self-shielding effect of the collapsed fuel was accounted for in the collapsed fuel region while the additional self-shielding effect in the lower nozzle region was conservatively neglected. The dose rate distribution from a 2x10 cask array was shown in Figure 5.
Table 3. Calculated packing fractions and associate fuel debris heights (from the bottom of the active fuel region) for 8x8 assemblies.

| Packing Fraction | Fuel Debris Height (cm) |
|------------------|------------------------|
| 0.3              | 291.181                |
| 0.5              | 168.412                |
| 0.63             | 130.412                |
| 0.75             | 107.028                |

5. DFC trail fabrication
The DFC weldment has a bottom plate that is 0.625 inches thick stainless steel. Four holes in the bottom plate, filter-screened with Type 304 stainless steel wire screens permit water to be drained from the can during loading operations. At the top of the DFC, the top flange extends beyond each of the four walls to allow the use of a handling tool to lift the can and the content in it.

The DFC walls consist of 0.05-inch thick Type 304 stainless steel sheet (18 gauge) plate at a total length of 177.57 inches. It is important to the forming precision as well to the welding process control. The DFC side wall is procured as sheet material and sheared to rectangular flat panels. The panels are then formed into “C” shaped sections to make a tube half. Two channels are fit-up, tack-welded and welded together with full-length longitudinal seams 180° apart with the laser beam welding (LBW) process.

6. Conclusions
The modification of the existing BWR basket design to accommodate both damaged fuel and high burn-up fuel presented the flexibility of the “tube-and-disk” design. The simple replacement of a fuel tube with the DFC in the 4 corners supported by minor design changes to the upper and lower weldments resulted in a very well performing design. As there was no increase in the heat loads, there were only minor thermal issues to address. NAC’s submittal also updated analytical methods where required due to the age of the original evaluations.

The NAC-UMS® is designed to store up to 56 BWR SNF assemblies. Up to four damaged BWR fuel assemblies may be loaded into a Class 5 damaged BWR fuel basket (BWR5 DF). Based on the length of the fuel assemblies, BWR fuels are grouped into two classes with different fuel lengths (Classes 4 and 5). Classes 4 and 5 BWR assemblies include the zircaloy channels. The SNF is loaded into a TSC.
which contains a stainless steel gridwork referred to as a basket. The applicant provided specifications for the higher enriched and higher burnup SNF in Technical Specifications for the new contents. The SAR sufficiently describes the radiation protection design bases and design criteria for the SSCs important to safety for the NAC-UMS® storage system. Radiation shielding features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 and therefore meets the design requirement of 10 CFR 72.236(d). The SAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

The design of the radiation protection system for the NAC-UMS® storage system with the additional low and high burnup BWR fuels is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the NAC-UMS® storage system will provide safe storage of SNF.

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