APPLICATION OF SCALE TO THE ANALYSIS OF SPENT FUEL CASKS

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Abstract

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The SCALE computational system was developed at the request of the United States Nuclear Regulatory Commission for use in cask license review and analysis. Of late, the system has been used in an increasing number of applications as its ease of use and versatility have become more widely known. The paper provides a brief review of the SCALE system and highlights various Oak Ridge National Laboratory applications of SCALE to illustrate the degree of standardization and simplification that the system can bring to design and review analyses of spent fuel casks.

1. INTRODUCTION

Since 1976, the U.S. Nuclear Regulatory Commission (NRC) has provided the Nuclear Engineering Applications Department at Oak Ridge National Laboratory (ORNL) with funds for development and maintenance of SCALE--a system which provides standardized analysis methods for nuclear fuel facility and package designs.[1] SCALE is a modular code system that enables a user to easily perform a variety of neutronic and thermal analyses by proper back-to-back execution of well-established functional modules. In addition, easy-to-use control modules have been developed to automate and standardize analytic sequences. Using a simplified, free-form input format, a user is able to prepare a control module input with easily visualized engineering parameters and keywords. The control module then automatically performs any necessary data processing (e.g., cross-section preparation), generates the input to the functional modules, initiates module execution in proper sequence, and performs any needed post-processing of the analytic results. Standardization is further enhanced by the incorporation of a host of validated data bases, e.g., material compositions, thermal properties, cross sections, which allow easy input (via keywords) and data accessibility.

The functional modules included in the latest publicly released version (SCALE-3) are: BONAMI, NITAWL, XSDRNPM, ICE, and COUPLE for cross-section and data processing; XSDRNPM, KENO IV, and KENO V.a for criticality safety analysis; XSDRNPM, XSDOSE, and MORSE-SGC for shielding analysis; ORIGEN-S for

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depletion analysis and evaluation of spent fuel radiation and heat source terms; HEATING6 for heat transfer analysis; and JUNEBUG-II, REGPLOT6, HEATPLOT, and PICTURE for plotting support. Control modules in SCALE-3 include: CSAS1, CSAS2, and CSAS4 for criticality safety analysis; SAS2 and SAS3 for radiation source term generation and shielding analysis; and HTAS1 for heat transfer analysis of a cask under normal and accident (transport) conditions.

New modules available for public release in 1987 include two new shielding analysis sequences for general one-dimensional (SAS1) and cask-specific multidimensional (SAS4) problems, a plotting package for ORIGEN-S, and an enhanced version of HEATING6 with multidimensional radiative heat transfer modeling capability (HEATING6.1). A simplified geometry input designed to allow either homogeneous or heterogeneous cask internals, automatic generation of Monte Carlo biasing parameters for the MORSE-SGC/S analysis, and default detector specifications make the SAS4 module a convenient tool for solving a very difficult deep penetration, three-dimensional shielding problem.

Recently, ORNL has employed SCALE in the following tasks: (1) independent review analysis of casks submitted for license approval, (2) study of radiation dose rates for current U.S. cask designs, (3) investigation of the potential advantages in allowing burnup credit for spent fuel casks, and (4) support of national (U.S.) and international (OECD) benchmark efforts. The application of SCALE in each of these tasks will be reviewed below. The emphasis will be on illustrating the versatility and convenience that the SCALE system brings to cask analysis. Note that the tasks were performed using the latest SCALE version available at ORNL, so several of the modules discussed are not yet publicly available. In presenting the applications, findings of significance to spent fuel cask design or analysis will be discussed.

2. INDEPENDENT REVIEW ANALYSIS

As noted in the Introduction, the chief motivation for developing SCALE was to provide an easy-to-use system for independent analysis of spent fuel cask designs. Recently, SCALE was used at ORNL to provide the U.S. Department of Energy (DOE) with an independent review of two dry transport/storage casks designed by Transnuclear (TN) for specific BWR and PWR fuel assemblies currently stored at West Valley, New York.[2] The ferritic steel casks will hold 85 "short" BWR assemblies and 40 standard PWR assemblies, respectively. The assemblies had relatively low burnup and cooling times of 10-12 years.

The thermal characteristics of the cask body under the NRC's normal and accident conditions of 10CFR71 were quickly verified using the HTAS1 module. The input for the HTAS1 module corresponding to the initial cask design with BWR fuel is shown in Fig. 1. The radius and half-height (in inches) of the cask cavity is specified on the CAVITY zone card along with the respective number of mesh intervals and the heat load (in Btu/hr).[1] The cask body (modeled as two concentric cylinders outside the cavity) is denoted by the INNER SHELL and OUTER SHELL zone cards containing the side, top, and bottom thicknesses (in inches). A MATERIAL index of 14 denotes the use of carbon steel property data. With the geometry specification provided, the automated HTAS1 sequence proceeds with an R-Z analysis for the cask using the

[1] 1 inch = 25.4 mm.
1 Btu = 1.055×10^3 J.

```
BWR CASK

CAVITY  32.00 10 85.5 15 20484
INNER SHELL  4.8125 10 4.25 5 4.125 5
MATERIAL 14
EMISSIVITY  5R0.66
OUTER SHELL  4.8125 10 4.25 5 4.125 5
MATERIAL 14
EMISSIVITY  3R0.6 2R0
PREFIRE  100 0.18 0.333 0.19 0.333 122.92
FIRE
POSTFIRE  180 100 0.18 0.333 0.19 0.333 0
%
END
```

FIG. 1. Example of HTAS1 input used for ferritic steel cask analysis.

HEATING6 code to obtain temperature results for a PREFIRE steady state, a 30 minute FIRE transient, and a 180 minute POSTFIRE transient. The output results showed good agreement with values submitted by TN.

In the shielding area, independent verification was obtained using the SAS2 and SAS1 control modules. The cask submittal called for the fuel to be grouped by burnup and decay time and then loaded into an inner and outer region of the cavity. Spent fuel radiation sources for each grouping were easily obtained with the depletion portion of the SAS2 control module. Results from the SAS2 cases also provided assurances that the specified maximum heat load was not exceeded. SAS2 utilizes ORIGEN-S for the isotopic depletion and decay and uses NITAWL-S and XSDRNPM-S to update the flux parameters and cross sections in an automated fashion during the depletion analysis. An output file containing source spectra (neutron and gamma) in an input-specified group structure is available from SAS2 or ORIGEN-S. Since the shielding portion of the SAS2 module is restricted to a one-dimensional (1-D) radial analysis and one source spectrum, the more general SAS1 module was used for the shielding verification. The SAS1 control module can read and use multiple source spectra files from SAS2 or ORIGEN-S and/or utilize a user-input source spectrum. Like SAS2, the SAS1 control module uses XSDRNPM-S to perform the 1-D radiation transport and the XSDOSE module to calculate the dose at a point exterior to the shield.

Input examples of SAS2 and SAS1 as used in the review are shown in Figs. 2-3. In order, the SAS2 input consists of 1) a sequence specification (=SAS2) and title card, 2) the cross-section library specification (27GROUPSHLD) and fuel geometry type (LATTICECELL) to be used for the depletion analysis, 3) the fresh fuel, reactor moderator, and cask material specifications, 4) the latticecell configuration (SQUAREPITCH) and geometric parameters, and 5) information for the fuel depletion and isotopic decay. The remaining input records specify the cross-section library, geometry, and fuel zones for the 1-D radial shielding analysis of the cask. However, the PARM=HALT02 on the first input card halted the SAS2 sequence after completion of the specified depletion and isotopic decay. The SAS1 input specifies 1) the sequence specification and title card, 2) the cross-section library and geometry (infinite homogeneous medium) for the resonance treatment, 3) material specifications, and 4) geometric specifications for the shielding analysis. Just prior to the READ XSDOSE

```
     PARM=HALT02
REG REACTOR, 14X14 PWR FUEL - 9.7 GWD/MTU, 33G/MTU CO, 13 YR. COOL
27GROUPSHLD   LATTICECELL
UO2  1 0.8854 1000 92234 0.03 92235 3.48 92238 96.49  END
EU-153   1 0 1-20 1000 END
EU-154   1 0 1-20 1000 END
ZIRCALLOY  2 1  605   END
BORON  3 1.482-4 591 END
H2O  3 0.6870 591 END
CO-59  3 0 1-20 591 END
FE  4  0  1-10  END
FE  5  0  5.804-2  END
CR  5  0  1.658-2  END
NI  5  0  1.064-2  END
B-10  5  0  1.4845-3  END
B-11  5  0  6.1013-3  END
N  6  0  5.4-5  END
CARBONSTEEL  7  END
END COMP
SQUAREPITCH  1.41224 0.93193 1 3 1.07188 2 0.94844 0  END
NPIN/ASSM=179  FUELNGTH=365.76
NCYCLES=2  NLIB/CYC=1  LIGHTEL=11
POWER=14.325  BURN=129.5  DOWN=0.01  END
POWER=14.325  BURN=129.5  DOWN=4800  END
SI  0.2  CR  4.8  O  51.4
MN  0.4  FE  14.4  CO  0.0128  NI  4.8
ZR  106.1  NB  0.3  MO  0.2  SN  1.7
27N-18COUPLE  TEMPCASK(K)=394.27 NUMZONES=4  END
4 72.9  5 73.8  6 91.1  7 114.6
ZONE=1  FUELBNDL=40
END
```

FIG. 2. Example of SAS2 input used for ferritic steel cask analysis.

card are six zone specification cards. The first and third are source zones (fourth entry = 1) which use SAS2 output source spectra from designated units (60 and 61). The conversion of the source from per assembly units to per cm^3 units and/or normalization is accomplished with an input zone volume and number of assemblies (last two entries on card). The default detector locations at 0, 1, 2, and 4m from the cask are used.

As exemplified by these examples, the HTAS1, SAS2, and SAS1 cases were easily and quickly set up using cask information from the safety analysis report. Thus, the use of SCALE enabled a timely and accurate independent review of the analysis supplied by the cask vendor. Although an example input is not shown, the CSAS25 sequence of SCALE was also used to review the criticality analysis submitted by the vendor. The cask model for the analysis included an explicit, heterogeneous description of each assembly using the array features available in the KENO V.a code. A number of criticality analyses were done per the NRC requirements of 10CFR71 to verify the optimum water density, study the sensitivity of k-eff to the boron weight percent in the basket, and evaluate various fuel loading configurations (for maximum k-eff). A loading accident scenario with an assembly lying on top of a full, open cask was also analyzed and shown not to affect the k-eff value. Of interest, however, was the fact

```
PWR CASK - 22 AT 11.5 GWD/MTU & 18 AT 9.7 GWD/MTU, 13 YR. COOLED
27N-18COUPLE  INFHOMMEDIUM
U-238    1 0 6.104-3  END
U-235    1 0 2.229-4  END
O        1 0 1.265-2  END
ZR       1 0 3.577-3  END
FE       2 0 5.804-2  END
CR       2 0 1.658-2  END
NI       2 0 1.064-2  END
B-10     2 0 1.484-3  END
N        4 0 5.4-5    END
CARBONSTEEL  5 1      END
END COMP
END
LAST
PWR CASK - 22 AT 11.5 GWD/MTU & 18 AT 9.7 GWD/MTU, 13 YR. COOLED
CYLINDRICAL
1 54.1 60 1 0 0 60 2 3.36311+6 22
2 54.8 1  0
1 73.4 25 2 0 0 61 2 2.74352+6 18
2 74.1 1  0
4 91.1 17 0
5 114.6 45 0  END ZONE
READ XSDOSE
457.2
END
```

FIG. 3. Example of SAS1 input used for ferritic steel cask analysis.

that for the cask with BWR fuel, the closed cask in water provided a 1% increase in reactivity over the open cask in water.

In a separate cask review project, the CSAS25 sequence was again employed to compare cask results with a heterogeneous, detailed assembly model and a homogenized fuel assembly model. Table I shows the k-eff results for three different storage casks and a typical shipping cask as analyzed with the 27 neutron group cross-section library developed for SCALE. The results indicate no discernible effect for PWR fuel. The single analysis performed with BWR fuel indicates there may be some small effect for BWR fuel, although a precise conclusion cannot be established with only one analytic result.

3. GENERIC STUDIES

The SCALE system has also been heavily used in performing generic studies related to spent fuel casks. One such study used SAS2 and SAS1 to obtain average radiation dose rates for consolidated and unconsolidated PWR spent fuel cooled beyond 5 years and loaded into current generation shipping casks.[3] A SAS2 case was run with a 17 x 17 assembly model to obtain a burnup of 33 GWD/MTU. The calculations were performed by stopping SAS2 following the depletion analysis, running ORIGEN-S stand-alone to the specified cooling times (5, 10, 15, 25 years) to generate radiation sources, and, finally, using these radiation sources as input to a SAS1 case or a SAS2 restart case with the appropriate cask model specifications. The ORIGEN-S

Table I. Comparison of k-eff values from heterogeneous and homogeneous cask models

	Fuel type	Effective multiplication factor, k-eff: Heterogeneous fuel model	Homogeneous fuel model
Storage cask #1	PWR	0.914 ± 0.005	0.919 ± 0.005
Storage cask #2	PWR	0.913 ± 0.004	0.909 ± 0.004
Storage cask #3	BWR	0.790 ± 0.005	0.809 ± 0.009
Shipping cask	PWR	0.883 ± 0.004	0.880 ± 0.005

output files differed for the consolidated and unconsolidated fuel in that Co-60 in the structural hardware of the assembly was removed for the consolidated fuel because the hardware is assumed to not be present in the cask and Co-60 contributes nearly all of the hardware's gamma source. The results indicated that a two-to-one fuel consolidation raises the surface dose rate over that obtained without consolidation anywhere from 30% (for 5-year-old fuel, rail cask with seven fuel cans) to 66% (25-year-old fuel, truck cask with one fuel can). If the hardware is included in the cask with the consolidated fuel pins, the dose rate could rise another 10-15%.

A comparison of cask dose rates obtained via Monte Carlo analysis and 1-D discrete ordinates analysis has been made with SCALE in an effort to assess the validity of using 1-D shielding calculations for casks. Results from SAS1 and SAS4 for radial dose rates along the axial centerline of a typical cask indicate reasonable agreement between the two methods. Also, buckling of the radial SAS1 case has minimal effect on the doses. However, later analysis of axial dose rates using SAS1 and MORSE-SGC/S indicates very poor agreement between the methods. Table II provides the comparative results. The SAS1 results overpredict the dose when no buckling is used and underpredict the dose when reasonable buckling values are employed. This effect is particularly dramatic for the neutron doses. So, although 1-D calculations can be useful for radial analysis of casks, caution is called for before they are used in axial evaluations of casks. This study provides excellent motivation for using the SAS4 sequence to analyze axial dose rates via the MORSE-SGC/S module.

Another study for which SCALE has a major role is a quantitative investigation of considering fissile inventory depletion and fission product buildup in the cask design process. For a 17 x 17 PWR assembly at 3.75 w/o U-235, SAS2 was used to obtain the depleted fuel isotopics at 5, 15, 18, and 33 GWD/MTU and at cooling times from 2 to 20 years. Only one SAS2 case for each burnup was required to generate the fuel isotopics and k_∞ values for the fuel pins at the required cooling times. The k_∞ values for the fuel pins obtained from SAS2 (via XSDRNPM-S) indicated a 2.8% reactivity drop for 18 GWD/MTU fuel as it decays from 2 to 20 years. For 33 GWD/MTU fuel, the reactivity drop was about 6.7%. In going from 18 to 33 GWD/MTU fuel, the reactivity drop was 10% for 2-year-old fuel and 15% for 20-year-old fuel. The SAS2 isotopics for the 18 GWD/MTU, 2-year-cooled spent fuel were selected as the conservative values to use for further study. Analysis of an infinite array of assemblies within a borated "basket" of 1.52 cm thickness (provided a multiplication factor slightly less than 0.95 for specified spent fuel) indicates that the fresh fuel equivalent for the spent fuel is 2.2-2.4% U-235. Further work in this area to assess the impact on future cask design is ongoing.

Table II. Calculated axial dose rates at canister surface for open IF-300/closed storage canister configuration

Analysis	Neutron dose[a] (mrem/hr)	Gamma dose (mrem/hr)
MORSE-SGC/S[b]		
R=0/20 cm	296.44(0.077)	
R=20/34 cm	254.15(0.057)	
R=34/42 cm	201.82(0.059)	
R=42/47.63 cm	189.89(0.066)	
R=0/34 cm	-	91.58(0.173)
SAS1 using XSDRNPM-S/XSDOSE[c]		
No buckling	1257.1	110.10
Cask radius-equivalent buckling	171.8	81.05
Canister radius-equivalent buckling	92.5	67.97

[a]Numbers in parentheses denote fractional standard deviations. 1 rem = 1.00×10^{-2} Sv.

[b]Inner/outer radii of circular detector area given.

[c]Dose rates at radial centerline of canister, R=0. Canister radius is 47 cm and cask radius is 63.02 cm.

4. COMPARATIVE BENCHMARK EFFORTS

Since the initiation of the SCALE project there has been an ongoing effort to validate the criticality codes, analytic sequences, and cross-section libraries against critical benchmark experiments relevant to casks and other applications. However, relatively little has been done in the areas of heat transfer and shielding, primarily due to lack of available data and/or benchmark problems. Recently, ORNL used the HEATING6.1 code with its enhanced radiation model to evaluate a set of U.S. benchmark problems developed by Sandia National Laboratories. The most severe test of the new radiation model was for a cylindrical prototypic cask with a voided neutron shield and a slab radiation shield located exterior to the cask. This was a good test of the multidimensional radiation capabilities in HEATING6.1. Temperature results, when compared with other thermal codes containing enhanced radiation models, agreed within 1-2% for the entire analysis scenario (steady-state, engulfing fire, and postfire).[4] Further efforts to validate the HEATING6.1 methodology for cask analysis is continuing via participation in an OECD international working group on heat transfer analysis of shipping casks.

The shielding analysis modules and cross-section libraries in SCALE are also being used to solve the benchmark problems developed for the OECD-sponsored international working group. In preparation for this effort, SAS1 analyses of a cast iron cask model were obtained using the 22n-18γ CASK library [1,5] and the 27n-18γ coupled SCALE library.[1] Table III indicates a significant difference in the neutron dose for the spent

Table III. Comparison of radial surface doses from a cast-iron cask model for two cross-section libraries[a]

	Dry cavity, 38-cm Fe shield		Dry cavity, 32-cm Fe with 6 cm polyethylene		H_2O filled cavity 38-cm Fe shield	
	27-18	22-18	27-18	22-18	27-18	22-18
Neutron	77.61	72.56	7.25	6.66	51.54	25.84
Primary γ	42.35	42.35	189.3	189.3	29.33	29.33
Secondary γ	0.51	0.51	5.78	4.49	0.28	0.16

[a]Doses are given in mrem/hr.

fuel cask containing water. The difference is primarily caused by a much larger fission source (more than doubled) in the case run with the 27n-18γ library. For the wet cavity case, the fission source is an order of magnitude larger than the fixed source, whereas the fission source is only about 20% of the fixed source for the dry cavity cases. Further efforts in the assessment and verification of available data libraries and methods for shielding cask analysis is continuing. The OECD work will also hopefully provide ample verification of the new SAS4 sequence.

5. SUMMARY

The SCALE system is an easy-to-use, well-documented computational system for performing cask analysis applicable to design, review, and generic investigations. New modules to be made available in 1987 will further enhance the capabilities of the system.

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