

SNF: SPENT FUEL ANALYSES BASED ON CASMO/SIMULATE IN-CORE FUEL MANAGEMENT

Axel Becker and Gerd Anton

Studsvik Scandpower GmbH

Norderstedt, Germany

axel.becker@studsvik.com, gerd.anton@studsvik.com

Sigurd Børresen

Studsvik Scandpower AS

Kjeller, Norway

sigurd.borresen@studsvik.com

Keywords: CASMO/SIMULATE, SNF, Spent Fuel Management, Decay Heat, Radiation Source Terms

ABSTRACT

This paper provides an overview of the SNF computer code¹ of the Studsvik code package. SNF has capabilities for back-end calculations consistent with, and utilizing in-core fuel management (ICFM), multi-dimensional analyses performed with e.g. the CASMO/SIMULATE system². SNF generates isotopic concentrations, decay heat and radiation source terms of fuel assemblies and/or individual fuel rods for cooling times up to 100,000 years. The basic methods of SNF, including some recent development is reviewed as well as examples of validation and application of the program.

1. INTRODUCTION

The number of filled fuel pools in the USA has already reached about 80% of the capacity and thus alternative storage has to be found. Some utilities use dry cask storage. Accurate prediction of the spent fuel characteristics is mandatory for safe and economic solutions and thus coupling of in-core and back-end methods appear advantageous.

In Germany SNF is in routine use by several utilities calculating fuel properties of discharged fuel assemblies for loading them into dry storage casks. Recently a new tool has been developed and applied in a realistic study optimizing the cask loading campaigns. After final shutdown of a reactor post-operation costs of a nuclear power plant are quite substantial as long as nuclear fuel is stored inside the reactor building. To empty the wet storage pool is a cost incentive of high importance. The cask loading optimization tool is of great help planning this process efficiently

2. SNF METHOD

Using CASMO/SIMULATE system, the fuel depletion is based on a nodal representation of fuel assemblies and includes also tracking of burnup distributions of individual fuel rods. Thus, at end-of-life (EOL) the ICFM database contains information

that may be utilized to reconstruct the isotopic contents in each axial node of a fuel assembly, and, if desired in each fuel rod. The SNF isotopic depletion model utilizes the accurate, local neutron spectrum based depletion calculation of the 2-D lattice physics code (CASMO or HELIOS) as well as nodal or rod-wise operating history data obtained from the 3-D simulator (e.g. SIMULATE). The simulator nodal power history is usually validated against plant data during the reactor operation. It is thus both accurate and very detailed, consisting of hundreds of time-steps.

2.1 SNF power history model

The SNF power history model accounts for the influence of the actual power density history on the isotopic concentrations generated by the lattice code at a constant power density. The EOL concentration of a given isotope in a given node is obtained by integration of the relevant isotopic buildup/decay chains through the entire in-core lifetime. Short-lived fission product isotopes are obviously very sensitive to the local power density, however, also isotopes generated by neutron capture in fission products and many actinides depend quite strongly on the power density history. An example of power history factors, C_p , of ^{241}Am and ^{242}Cm for a 5-steps test case is shown in Fig. 2.1. The C_p -factor is defined as the isotope concentration integrated over the variable (actual) power density history relative to the concentration based on a constant power density. Reference results generated by running the 2-D lattice code with the variable power history are shown for comparison.

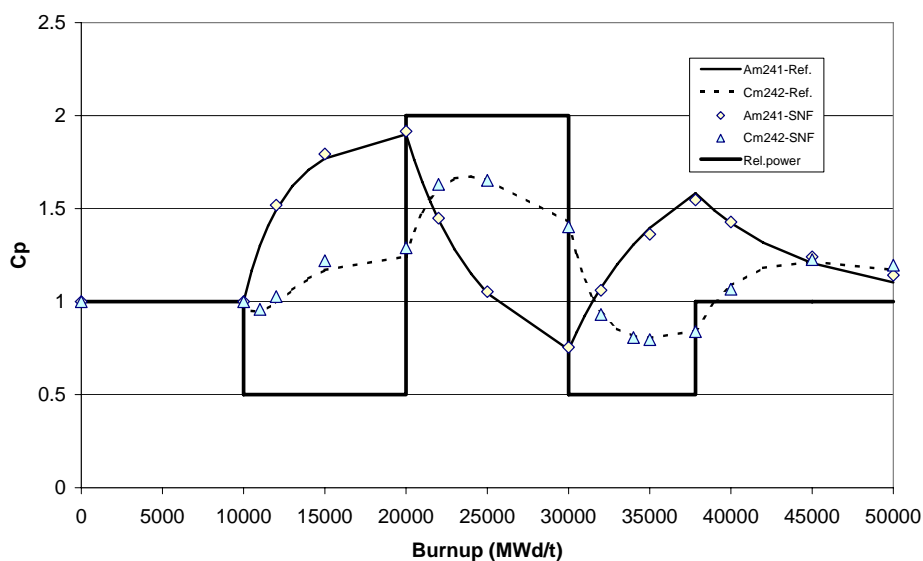


Fig. 2.1 Example of SNF power density history correction factors for ^{241}Am and ^{242}Cm for a 5-steps test case.

2.2 Isotopic decay and radiation sources

The isotopic concentrations at discharge are used as initial conditions for solving the isotopic decay chains. A number of isotopes not present in the lattice physics codes are added in this process. The final isotopic concentrations provide the required basis for

calculation of radioactivity, decay heat, gamma heat, spontaneous fission and (α, n) -reaction source neutrons as well as photon release rates and spectra. The SNF library data are, for the most part, based on the ENDF/B-VI Decay Data File.

The photon spectrum is based on decay gammas and x-ray data from the ENDF/B-VI file and includes also models for photons associated with spontaneous fission and (α, n) -reactions of actinides and bremsstrahlung from the most important β -emitters.

SNF applications include analyses of radioactivity, decay heat and neutron emission rates of spent fuel assemblies to be loaded into transport/storage casks as well as full core or fuel pool decay heat calculations required for demonstration of compliance with cooling capacity limitations³. The latter function has been integrated into an ‘on-line’ module (DHM: decay heat module) available to the station physicist.

2.3 Short cooling times

For cooling times less than a few days, many short-lived fission products that are not present in the 2-D lattice physics libraries, and hence not available in SNF, are important for proper calculation of e.g. the decay heat. These fission products are represented by a ‘lumped’ fission product model in SNF. The ANS decay heat standard is utilized to obtain the decay heat of the lumped fission products. However, all important short-lived actinides, and also isotopes generated by neutron capture in fission products (e.g. ¹³⁴Cs) are explicitly represented in SNF, also for short cooling times.

3. RECENT DEVELOPMENT

3.1 Fuel rod calculations

A method for prediction of isotopic concentrations and radiation source terms of selected fuel rods, or in ‘samples’ at given axial locations has been added as a new calculation option of SNF. An example of rod-to-assembly isotopic ratios for a low enriched corner rod of a spent BWR fuel assembly is shown in Fig. 3.1.

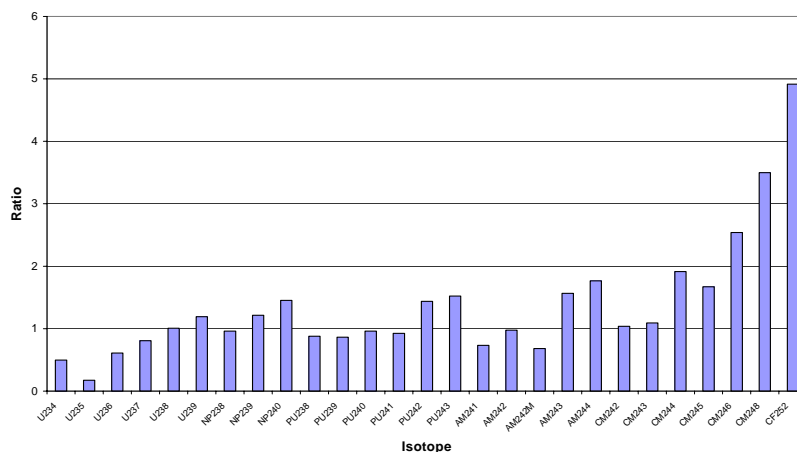


Fig. 3.1 Example of rod-to-assembly isotopic concentration ratios, BWR corner rod.

As may be seen, the local concentrations differ considerably from the assembly average ones.

Fig. 3.2 shows the rod-to-assembly ratios of the specific activity, decay heat and neutron source of the corner rod vs. cooling time. The activity and decay heat of the corner rod are not very different from the assembly values (<10%), however, the specific neutron source is nearly twice that of the assembly. This is consistent with the very high concentrations of ^{244}Cm , ^{246}Cm and ^{252}Cf shown in Fig. 3.1.

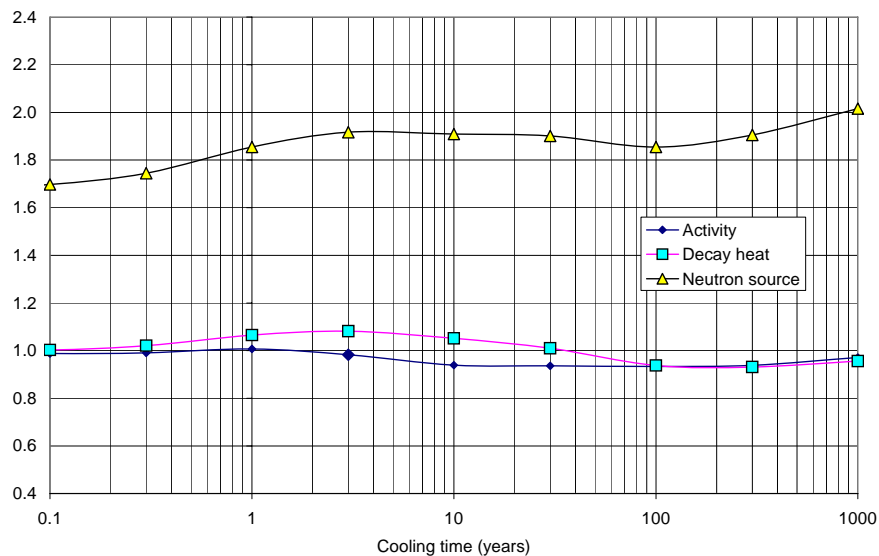


Fig. 3.2 Rod-to-assembly ratios of activity, decay heat and neutron source, BWR corner rod.

3.2 Helium accumulation

Another new feature is a model for detailed tracking of the helium buildup during irradiation and decay that is of special interest in studies of the long-term evolution of spent fuel. This is of particular interest for analyses of MOX fuel that generates considerably more helium than uranium fuel (cf. Fig. 3.3 below).

The build-up of helium during the irradiation of the fuel in the reactor is mainly caused by alpha-decay of ^{242}Cm and ^{244}Cm . These isotopes contribute typically 90% of the total alpha-decay. The plutonium isotopes are also of some importance and there is a small contribution from ternary fission. A ternary fission yield for helium of $2.0 \cdot 10^{-4}$ is used in SNF. The accumulated helium concentration at EOL is obtained by integration of the isotopic alpha-release rates over the in-core life-time. Due to the importance of the curium isotopes their dependence on the power density history has been taken into account in the integration model (cf. Fig. 2.1). The additional helium build-up during the decay period is obtained by numerical integration of the total alpha-release rate from EOL to a given, final cooling time. An example of SNF-calculated helium build-up in UO_2 and MOX fuel is shown in Fig. 3.3.

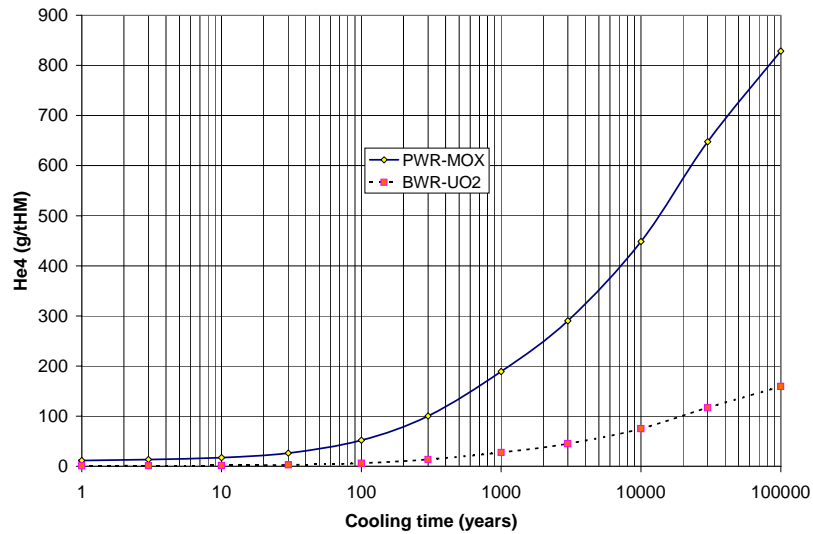


Fig. 3.3 Example of helium accumulation vs. cooling time (burnup 40 GWd/t)

4. VALIDATION

SNF has been extensively validated against experimental data, decay heat standards and other calculations. The isotopic decay, source terms and photon spectra have been validated by comparison with ORIGEN-ARP decay calculations, starting from identical isotopic concentrations at discharge. Such comparisons show agreement within a few tenths of a percent in total activity, decay heat and neutron source rates.

An example of decay heat comparisons for actinides at 100,000 years cooling time is shown in Fig. 4.1.

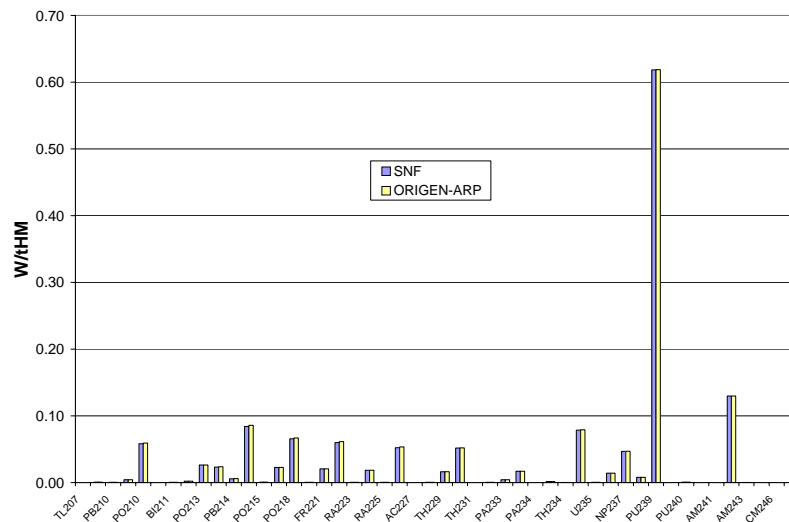


Fig. 4.1 Comparison of actinides decay heat at 100,000 years cooling time - SNF vs. ORIGEN-ARP

The most relevant validation of SNF consists of comparisons with experimental data of decay heat and neutron emission rates of spent BWR and PWR fuel assemblies⁴. Decay heat comparisons with experimental decay heat data of 16 BWR and 33 PWR spent fuel assemblies from Swedish reactors are shown in Fig. 4.2. The average C/E-value of this comparison was 0.994 ± 0.019 . The BWR assemblies included both traditional 8x8/9x9 designs as well as water-cross assemblies. A comparison of the primary neutron emission rates of the same 16 BWR fuel assemblies with the Würz experimental correlation⁶ for 8x8 BWR fuel is shown in Fig. 4.3. The average C/E-value of the 8x8/9x9 fuel assemblies was 0.98 ± 0.06 , whereas the water-cross assemblies fell a little below the experimental correlation, as should be expected due to the softer neutron spectrum.

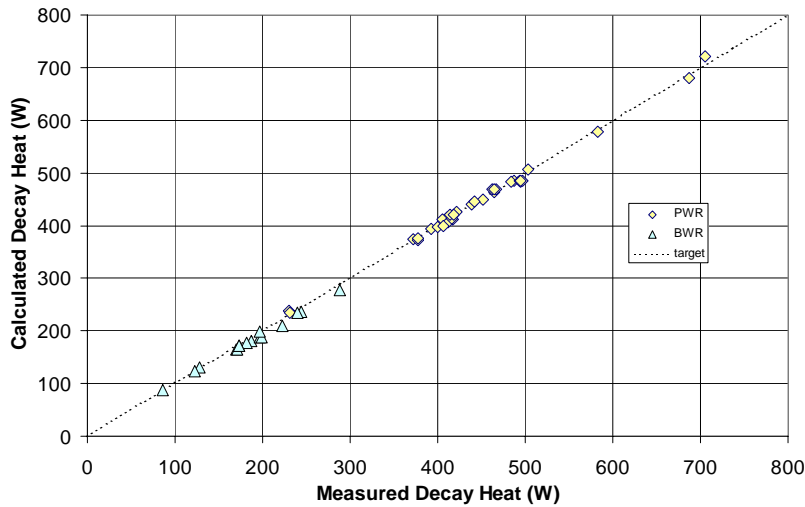


Fig. 4.2 Calculated vs. measured decay heat of 49 BWR and PWR spent fuel assemblies

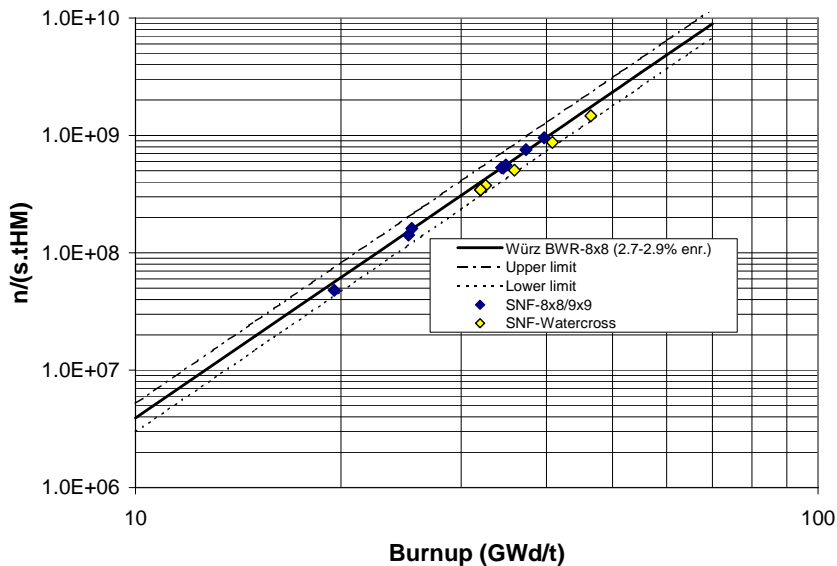


Fig. 4.3 SNF vs. Würz neutron emission correlation for BWR spent fuel assemblies.

The good agreement obtained between the calculated and experimental neutron emission indicates that the ^{244}Cm concentrations are well predicted. This is somewhat contradictory to isotopic assay comparisons, both with CASMO-4, HELIOS and other lattice physics codes that often show under-predictions of typically 20% of the curium isotopes¹. However, in the overall evaluation of SNF we tend to rely more on the neutron emission results and thus do not recommend any artificial correction of the theoretical curium contents. This is further supported by the good agreement of the decay heat comparisons where ^{244}Cm contributes up to about 20% of the calculated values.

5. APPLICATION

SNF is in routine use for German reactors calculating fuel properties of discharged fuel assemblies for cask loadings. Cask design criteria are a maximum gamma dose rate of 0.1 mSv/h and a maximum neutron dose rate of 0.25 mSv/h at the cask surface. In addition, a limit for total decay heat of a cask is given. To meet these design criteria several limiting parameters are defined for fuel assemblies to be accepted for a cask loading, in addition to the decay heat: S1 is a parameter restricting the gamma source term, S2 is a limit for the neutron source term, and S is a combination of S1 and S2 relevant for cask transportation.

$$S1 = \sum_{i=1}^7 \frac{G_i}{G_{i0}} \quad S2 = \sum_{j=1}^2 \frac{N_j}{N_{j0}} \quad \begin{array}{l} G_i, N_j - \text{gamma, neutron source terms} \\ G_{i0}, N_{j0} - \text{reference source terms} \\ i, j - \text{energy groups} \end{array}$$

Fig. 5.1 S1 and S2 definitions

Recently a study has been performed for a German BWR planning the cask loading campaign to empty the fuel pool completely. This reactor is supposed to be shutdown finally after two cycles from now. As long as fuel is stored in the fuel pool the operational costs of the plant are quite substantial. To reduce the reactor post-operation cost the goal is to load all fuel assemblies into casks a.s.a.p. after reactor shutdown. The study comprises three main steps:

- simulating future cycles operation,
- calculating properties of the discharged fuel assemblies,
- and planning cask loadings to empty the fuel pool inside the reactor as early as possible.

The last two cycles are designed to achieve the desired energy output. Fuel assembly loading patterns are developed using the numerical BWR optimization code PARAT⁶. Cycle operations are simulated as realistic as possible, e.g. modeling spectral shift and stretch out operation explicitly. Thermal limits and shutdown margin have to meet the required design limits. Fig. 5.2 is illustrating the planned cycle operation showing reactor power and MFLCPR as examples. The main results for further investigation of the post-operation phase are predicted nodal burnup distributions and isotopic concentrations of all discharged fuel assemblies.

Discharged fuel assembly properties like decay heat, neutron source etc. are calculated by SNF with the same nodal resolution as the burnup distribution and the isotopic concentration derived by the in-core fuel management tools. The methods for in-core fuel management and back-end calculations are fully consistent.

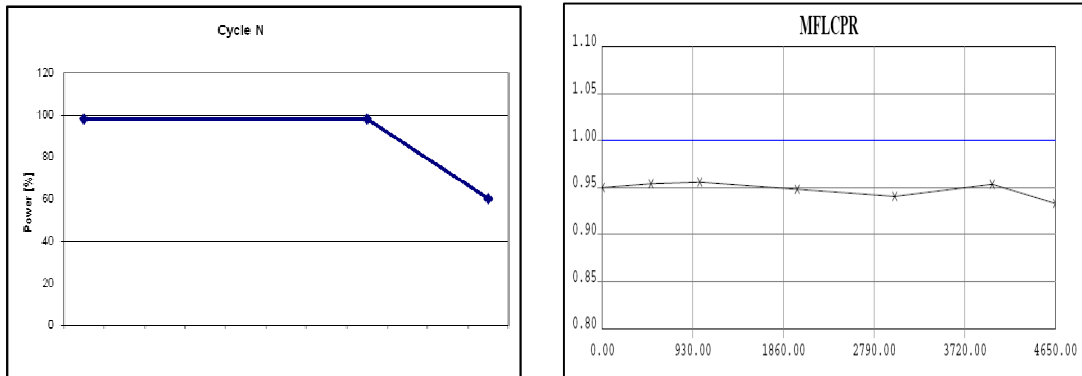


Fig. 5.2 Reactor power and thermal limit MFLCPR of cycle N

Based on accurate properties of the discharged fuel assemblies cask acceptance criteria are calculated as a function of time. Different cask loading types (Fig. 5.3) are available having different fuel assembly acceptance criteria depending on the loading type: homogeneous or heterogeneous loading and 52 or 32 assemblies loading. Within the loading types different fuel zones are defined. In Fig. 5.3 the fuel loading zones are marked as ‘U’ and ‘S’ positions.

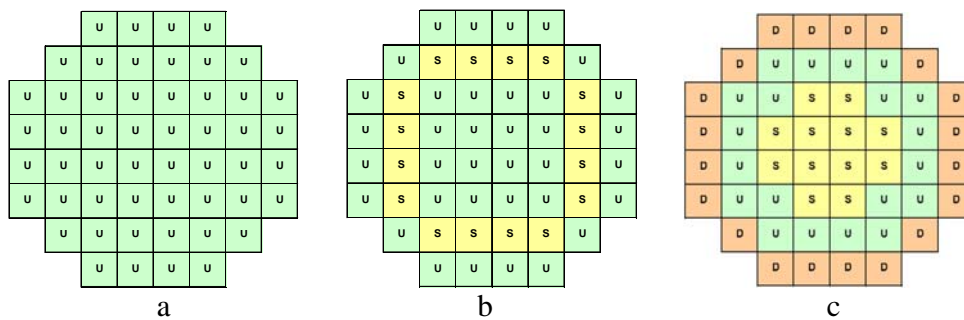


Fig. 5.3 Examples of cask loading types: a) homogeneous loading pattern with 52 and heterogeneous loading patterns for b) 52 and c) 32 locations

‘U’ locations are appropriate for uranium bundles up to 55 GWd/tU, ‘S’ locations can be used to load bundles with a burnup up to 65 GWd/tU. ‘S’ locations can accept higher decay heat as well as higher radiation compared to ‘U’ locations. The 32 bundle loading type contains 20 so-called dummy assemblies (‘D’) at the periphery. This means

that a 32 assembly loading allows higher values compared to the 52 loadings. Therefore, the ‘S’ locations of the 32 cask type are providing the highest limits for fuel assemblies.

Altogether 728 discharged fuel assemblies are analyzed calculating the cooling times needed to meet the different cask loading criteria. The final step of the study is assigning each fuel assembly to a certain loading date and a specific cask loading zone. This task is quite complex because of the number of possible combinations. A new program, CASKLOAD⁷, developed in cooperation with a utility has been applied to this problem as a pilot project. The tool optimizes the whole loading process. First of all, the loading dates and the number of casks available at each point in time have to be defined. The code tries to find a date and a loading zone for each assembly to utilize the cask capacities in an optimal way. The goal function of the code is defined to minimize the total costs for the reactor post-operation and the investment for casks needed to load all fuel assemblies completely.

Fig. 5.4 shows the number of fuel assemblies fulfilling the cask loading criteria for the ‘S’ locations of a 32 or a 52 bundle cask. 37 months after final reactor shutdown all fuel assemblies meet at least the criterion for the ‘S’ location of the 32 cask loading. This is the first feasible point in time when it would be theoretically possible to empty the reactor pool (c.f. Fig 5.4: X – reference point in time). The criterion of an ‘S’ position in a 52 assembly loading is fulfilled by all assemblies after 51 months. This is the minimal cooling time needed in case only 52 fuel assembly loadings would be used. In this case the minimum number of casks needed to load all 728 fuel bundles is 14.

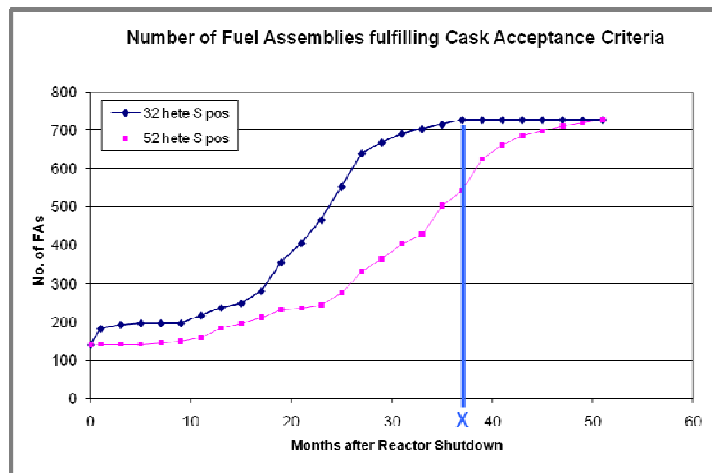


Fig. 5.4 No. of fuel bundles fulfilling the 32 or 52 cask loading criteria

Based on SNF calculated decay heat and source term parameters the cask loading strategy is developed taking into account additional boundary conditions and assumptions like loading dates and available number of casks at each date, time needed to load one cask and other restrictions like capacity of the fuel storage pool. Finally, the price of a

cask and the reactor post-operation costs per months are the basis to find the optimal solution giving the minimal total costs.

Different scenarios are analyzed. A first theoretical scenario is analyzed trying to load all the bundles exactly 37 months after reactor shutdown. The code finds a solution using 21 casks: 16 heterogeneous loadings with 32 assemblies and 5 heterogeneous loadings with 52 assemblies.

In reality the loading of one cask takes about 1-2 weeks. Therefore, two scenarios are analyzed as lower and upper limits: scenario A takes one week as time needed for one loading, scenario B takes two weeks. In addition, one extra week after 4 loadings is defined as a time reserve.

Scenario A needs 20 casks to load all fuel assemblies, 15 heterogeneous loadings with 32 assemblies and 5 heterogeneous loadings with 52 assemblies. X is taken as the reference point in time when all fuel assemblies meet the loading criterion of the 32 ‘S’ locations (37 months after final reactor shutdown). The first cask can be loaded 11 weeks before the reference date and the loading campaign is finished about 13 weeks after reference date. Fig. 5.5 shows the details of the loading schedule.

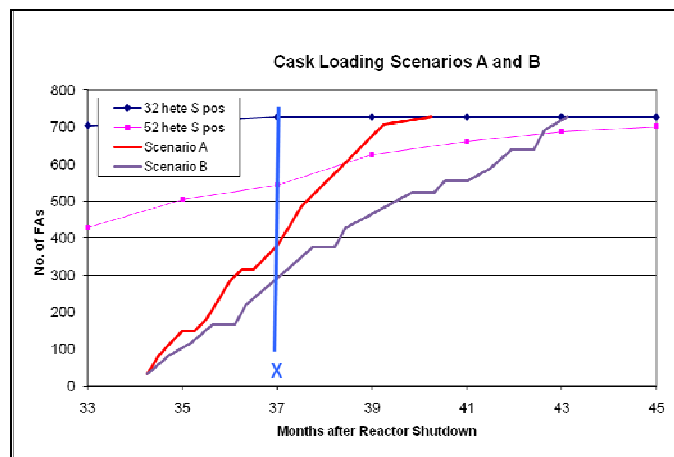


Fig. 5.5 Cask loadings of scenario A and B, X is the reference point in time when all fuel assemblies meet the ‘S’ criterion of the 32 loading

Scenario B needs 17 casks to load all fuel assemblies, 7 heterogeneous 32 bundle loadings and 5 heterogeneous 52 loadings. The loading campaign starts the same way as in scenario A but takes about 3 months longer than scenario A. The longer cooling time saves 3 casks compared to scenario A. But the post-operation costs per month for the plant are about three times higher than the investment for one cask. Therefore, scenario A is giving the minimal total costs.

Another interesting consideration is to analyze how the capacity of the casks are utilized. The table (Fig. 5.5) shows for the decay heat maxima the percentage of the limits of U and S locations as well as the total sum relative to the total capacity of the U and S regions. The maximum allowed decay heat is utilized in almost all casks whereas

the total sum reaches the maxima mainly for the first and last loading for the S locations. The evaluated cask loading plan obviously is utilizing the cask capacities to a very high extend which proves the functionality of the code. In average this loading campaign reaches about 75% of the cask capacity.

Cask No.	Cask Type	Max/Limit in U	Max/Limit in S	Sum/Limit in U	Sum/Limit in S	Sum/Limit in (U+S)
1	32	83%	99%	52%	92%	73%
2	52	54%	100%	52%	98%	69%
3	32	72%	97%	51%	91%	72%
4	52	100%	100%	60%	98%	74%
5	52	50%	100%	42%	100%	64%
6	52	100%	99%	70%	93%	78%
7	52	100%	100%	76%	78%	76%
8	52	100%	100%	86%	74%	82%
9	52	99%	100%	79%	99%	86%
10	32	99%	90%	65%	88%	77%
11	32	67%	80%	40%	66%	53%
12	32	100%	65%	58%	61%	59%
13	32	69%	64%	53%	61%	57%
14	32	100%	61%	100%	58%	78%
15	52	100%	99%	96%	95%	96%
16	52	99%	100%	86%	99%	91%
17	52	71%	95%	27%	89%	50%

Fig. 5.5 Cask utilization of the maximum and average capacity in percent

This study provides a very precise basis for technical planning of cask loadings and as well as a founded basis for the financial considerations involved in the back-end planning process.

6. SUMMARY AND CONCLUSION

SNF is directly coupled to CASMO-SIMULATE making use of the detailed nodal representation of the in-core fuel management (ICFM) core tracking results when calculating discharged fuel properties. ICFM and SNF methods are fully consistent. The same 3D nodal resolution is applied in SNF taking into account the detailed nodal power history. The latest SNF version is even capable to treat single fuel pins for special applications. The approach integrating SNF with CASMO-SIMULATE provides a code package which is easy to use and which assures together with the extensive validation base high accuracy of the results.

SNF is in routine use by several utilities today. The code package has been extended recently for cask loading planning. The new program CASKLOAD demonstrated its capability to plan and to optimize a cask loading campaign successfully. Based on a reactor operation prediction until end of life a cask loading campaign has been planned covering all discharged fuel assemblies until end of reactor life. Altogether this integrated code package is an excellent tool set to analyze cask loading strategies to find the optimal loading campaign under given conditions and restrictions.

REFERENCES.

1. SIGURD BØRRESEN, “Spent Nuclear Fuel Analyses based on In-Core Fuel Management Calculations”, The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments, PHYSOR 2004, Chicago, IL, USA, April 25-29, 2004
2. M. EDENIUS, K. EKBERG, B.H. FORSSEN, AND D. KNOTT, “CASMO-4: A Fuel Assembly Burnup Program User’s Manual”, Studsvik/SOA-95/1, Studsvik, (1995).
3. AXEL BECKER, GERD ANTON, AND SIGURD BØRRESEN, “SNF Calculations of Spent Fuel Source Terms for Fuel Pools and Storage Casks”, 2007 ANS/ENS International Meeting, Washington D.C., USA, November 11-15, 2007.
4. SIGURD BØRRESEN, AXEL BECKER, “Calculation of neutron emission from spent LWR fuel assemblies: SNF method and validation”, International Conference on Reactor Physics, Nuclear Power: A Sustainable Resource, Interlaken, Switzerland, September 14-19, 2008.
5. WÜRZ, H., “A Simple Nondestructive Measurement System for Spent-fuel Management”, Nuclear Technology, pp 193- 206, Vol. 95 Aug. 1991.
6. A. BECKER, G. ANTON, J. AXMANN AND W. SCHÄFER, “Erfahrungen mit der numerischen Beladeplanoptimierung für Siedewasserreaktoren mit PARAT-BWR“, Annual Meeting on Nuclear Technology, Dresden, Germany, May 15-17, 2001.
7. A. FERRI, “CASKLOAD User Manual“, MAN-2006-0501, rev.2, Jan. 2008.