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DPT PROJECT DESIGNING A DUAL PURPOSE CASK FOR TRILLO NPP

J. M. GRAVALOS

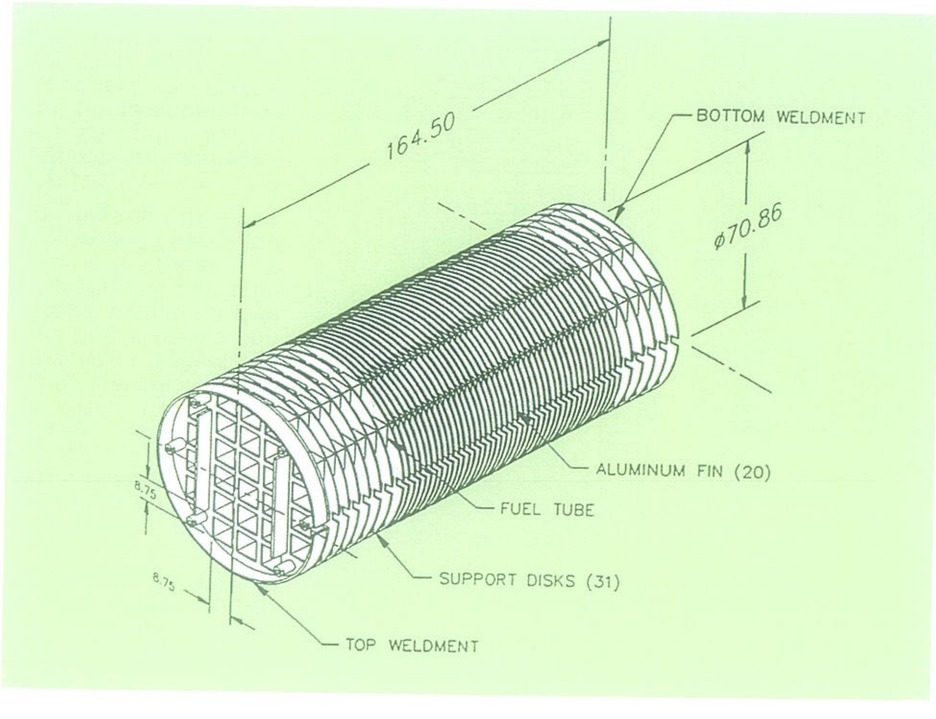
GENERAL

The Nuclear Regulatory Commission (NRC) has issued to the US company NAC Services Inc. a Compliance Certificate (COC) for the dual-purpose NAC-STC cask. This certificate, equivalent to a transport license, represents the first of the two licensing approvals required for the NAC-STC to become the first dual-purpose cask licensed by the US Regulatory Body. According to NAC sources, the NRC is expected to issue its approval for the cask to be used for the storage of spent fuel during the last quarter of 1994. Design of this cask has been promoted by ENRESA with support from the Electric Power Research Institute (EPRI) and the US electricity utility Virginia Power. The companies Equipos Nucleares, S.A. and Empresarios Agrupados have also participated in development.

This milestone is an important step in development of a project that ENRESA began in 1990, the aim of which is to make available in Spain casks allowing the storage capacity of NPP sites to be increased in case the spent fuel pool re-racking process were insufficient. The casks also provide ENRESA with the fuel handling capacity required to respond to the eventual needs of the different plants.

One of the reasons underlying the choice of these casks as a storage technology is their independence, such that once loaded at the nuclear power plants they may be stored on site until the decision is taken to transport them elsewhere (temporary centralized storage facility, deep geological disposal facility), without the need for the fuel to return to the storage pools and, therefore, without disturbing the normal operation of the plant. A second reason is their flexibility, since their modular design allows them to be manufactured and supplied according to the needs of the different nuclear plants, with regard to both the delivery schedule and technical aspects, and fundamentally to different fuel types. Finally, the possibility of incorporating this technology into the national industry was also a determining factor; in this respect the project was organized in such a way that technology transfer be maximized and that the design and manufacturing of the casks might in the future be undertaken from an eminently national standpoint.

F I



Manufacturer	Type Included	Comment
WESTINGHOUSE	14 x 14	Inside envelope
	15 x 15	Criticality Source Term
	17 x 17	Fuel Source Term (Shielding)
COMBUSTION ENGINEERING	14 x 14	Inside envelope
	15 x 15	Inside envelope
	17 x 17	Inside envelope
BABCOCK & WILCOX	15 x 15	Inside envelope
	17 x 17	Inside envelope
Total Heat Load	22.1 kW	

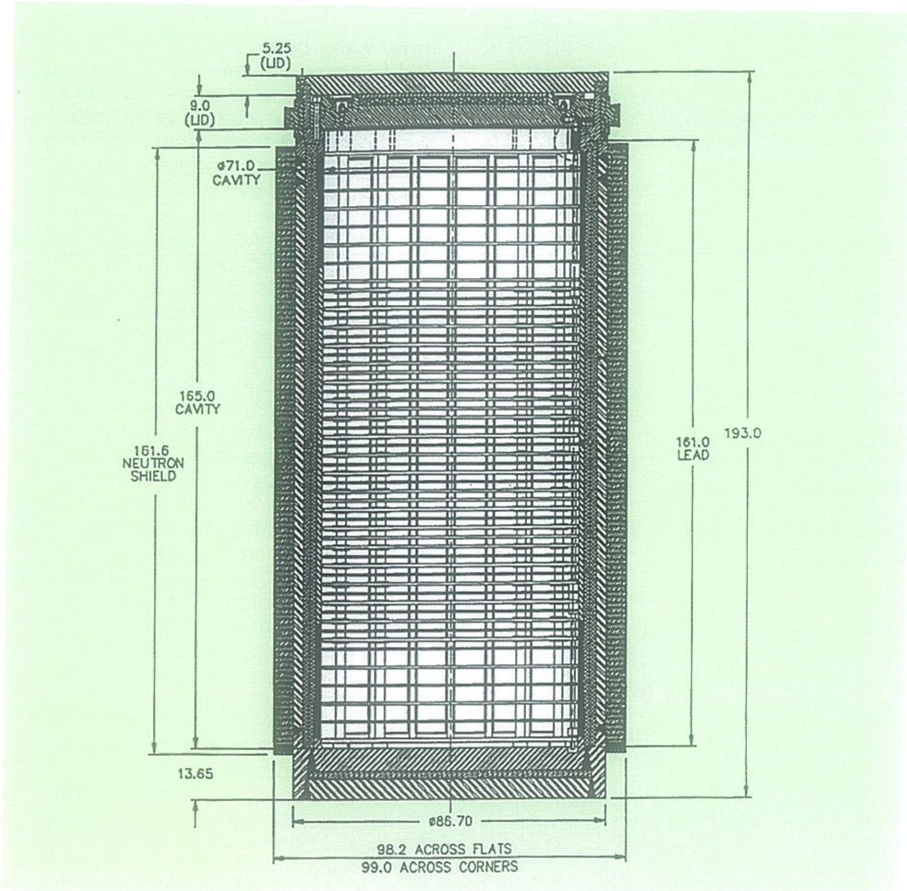
40,000 MWD/MTU and 6.5-year cool time
or
45,000 MWD/MTU and 10-year cool time

F II *Fuel type envelope NAC-STC design basis*

Licensing of this cask is being carried out by the NRC in the United States since it originated in that country and has potential users there. In Spain it will be necessary to obtain a license from the Nuclear Safety Council (CSN) and the Ministry of Industry for the cask to be used for storage; as regards its use for transport, the only requirement will be its inclusion on the register of casks approved for this purpose, following homologation as a B(U) type package. Detailed tracking of the licensing process has in itself constituted a transfer of technology, since it has allowed ENRESA and its contractors to gain detailed insight into the process applied by the NRC and prepare for licensing in Spain of this and subsequent casks, to be developed depending on the needs of the nuclear power plants.
As a result of this technology transfer, development of the next dual-purpose cask, in this case designed specifically for the Trillo NPP, is being undertaken by the company Equipos Nucleares, S.A., which, with collaboration by NAC, has recently completed the corresponding licensing documentation and submitted it to the Spanish nuclear authorities for evaluation.
What follows describes the main characteristics of the cask and the calculation tools used in its design, along with a brief summary of certain aspects of the licensing process carried out by the US NRC, in relation especially to certain unique aspects of a process which has simultaneously covered transport and storage and which has required specific documentation to be drawn up for both objectives.

CASK DESIGN

Nuclear Assurance Corporation (NAC) has designed and tested a dual purpose cask, shown in Figure I, for spent fuel known as the NAC-STC. The design and test results are described in an extensive and detailed Safety Analysis Report (SAR) submitted as part of the application for a USNRC transport Certificate of Compliance in accordance with 10 CFR 71, and an equally extensive Topical Safety Analysis Report (TSAR) for the companion USNRC storage license under 10 CFR 72.
The design basis spent fuel that can be accommodated in the STC is high-performance fuel of the PWR type. The fuel parameters can be as high as 4.2%. U235 initial enrichment, a residual heat content as high as 45,000 MWD/MTU, and a decay time as short as 6.5 years. A total of 26 PWR assemblies can be accommodated for each of the fuel types shown in Figure II. As shown, different fuel types were the envelopes for different conditions. Meeting all of these diverse and, to some degree, conflicting licensing requirements has resulted in some unique analyses and a creative use of proven (that is USNRC accepted) materials. In addition, the weight of the fully loaded package and its accessories has been maintained at less that 114 t for all handling conditions.
The STC is qualified for use in two types of scenarios. One is for storage of spent fuel at a reactor site for an extended period of years, followed by transport to any other site, where it may again be stored for an extended period. Alternatively, in the second scenario, the loaded STC may be shipped immediately, simply as a licensed transport cask, and used multiple times in numerous transport campaigns over its lifetime. Obviously, there are also any number of scenarios that combine the two above, expanding the STC's versatility. This versatility also results from unique operational features that meet all of the requirements for storage and transport. Principally, this relates to the ability to accommodate extended storage, followed by transport, without the need to unload the spent fuel or otherwise handle it. Demonstration of containment integrity is, of course, the key in this regard. The containment is defined as the inner cask body shell and associated end forgings; the inner lid with its double metallic O-rings; and the various ports with their cover plates, each with their test ports and their associated double metallic O-rings. In addition, there is a testable interlid region between the two lids that represents a cover volume over the potential leakage paths of the containment. Subsequent to extended storage, and prior to transport, the containment integrity can be reverified, as it can be after transport if the cask is to be stored again rather than unloaded (such as, for example, at a Centralized Interim Storage). Each set of metallic O-rings form an interseal volume that can be leak checked to demonstrate O-ring integrity. The interlid region is initially filled with helium at the start of any planned storage. This region is at one atmosphere pressure above the cask internal cavity pressure which was also filled with helium at the onset of storage. Periodic checking of the transducer attached to this interlid region demonstrates continued leaktight integrity of this region and, indirectly, the cask cavity integrity due to the interlid overpressure. At the end of the storage period and prior to transport, the interlid volume is sampled to detect the absence of any radioactive materials. With this accomplished, the outer lid is removed and the cask containment reverified by reinserting clean helium into the inner lid interseal volume and checking its leakage. Each port cover's interseal volume is also refilled and leak checked; and finally, the interlid volume is refilled and leak checked. In this way, containment integrity is reestablished prior to transport.
The validation test program, conducted to meet the USNRC hypothetical transport accident requirements, was quite extensive. A series of 13 quarter-scale model drop tests was conducted using different impact limiter designs and different pins to demonstrate energy absorption and puncture resistance. An analytical computer model was developed from model impact limiter tests and limited materials crush tests. This model, called RBCUBED, was used to predict cask deceleration and loads prior to conducting the drop tests. As can be seen, the predictions and test results are in good agreement. In addition, tests showed that the design basis decelerations were not exceeded. One particular drop test subjected the NAC-STC model to acceleration loads five times the design loads, demonstrating the conservatism in the package design.
The analytical tools developed for the design of the NAC-STC are extensive and extend the state of the technology in some cases. For example, the thermal model used for fuel and package material temperature prediction contains 49,000 degrees of freedom, including multiple-gap conductivities and radiation heat transfer modelling. The structural analyses used ANSYS finite element modelling with similarly detailed nodes



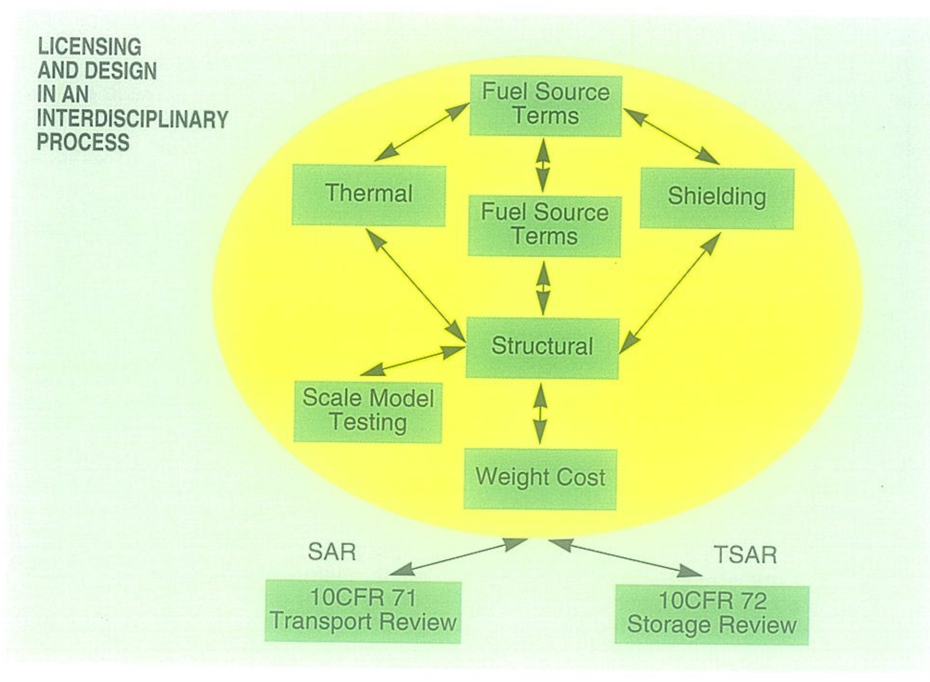
F III

considered. In the nuclear criticality analysis, the package must remain subcritical under a range of moderator densities and package spacings. Maintenance of the basket/fuel assembly configuration and neutron absorber loading are crucial to criticality safety. The analysis effort required approximately 50 KENO-VA Monte Carlo runs. Each run simulates neutron multiplication by tracking 500 neutrons per generation for 100 generations. Thus, approximately 2.5 million neutrons are tracked through the criticality model. In the shielding analysis, both finite difference transport theory (XSDRNPM) and Monte Carlo (MORSE) are used. In the Monte Carlo analysis, approximately 12 million neutron and gammas are tracked through the shielding model.

LICENSING PROCESS

The USNRC review of the overall package design has been quite extensive. A total of 600 questions were asked 60 percent in the structural or thermal area; 30 percent in the nuclear area—i.e., criticality or shielding; and the remainder were operational or manufacturing related. These questions, of course, came from the USNRC transport group review and from Lawrence Livermore National Laboratory acting as a consultant to the USNRC storage group. The most difficult part of the technical review centered around the design details of the basket. Basically, the cask body received an early review and acceptance—i.e., the stainless steel multi-wall and lead part of the package. This is the more traditional, proven technology and, so, was the subject of minimum new issues. The basket, however, received intense scrutiny and resulted in several significant modifications prior to acceptance by the USNRC. The resulting design shown in Figure III may be described as a tube and disk concept where the disks are either stainless steel or aluminum, with a total of 51 disks (31 steel and 20 aluminum). The steel disks are ASME Code approved material 17-4 PH stainless steel and are the key structural members that support the fuel in any hypothetical accident scenario applying code stress rules. The aluminum disks are used purely for their superior heat transfer properties and light weight, performing no structural function. These aluminum members are actually referred to as fins to distinguish them from the steel structural members. The exact spacing of the steel disks was a subject of some discussion with the USNRC. The concern here was related to the use of tubes containing BORAL (which is not a Code material) and the desire to be certain that the fuel load distribution was uniformly transmitted to the cask body in hypothetical accident conditions. Additionally, there is a need to consider the wide variety of possible fuel contents in the cask, each with their unique grid spacing, and the possibility of off-center alignment of the fuel grids and the disks. Thus, the spacing of the steel support disks was the key dimension to be selected. The dimension ultimately was selected assured that structural loads during accident scenarios, as a result of assumed fuel impact on the tubes, would be uniformly transmitted to the cask body.

F IV *Cask design analysis*



The tubes of the basket use poisoned materials to met the fresh fuel criticality requirement (i.e, no burnup credit). The tubes are not attached to the disks or fins, but are free to expand independently of the disks or fins. Nevertheless, assurance that the disks would be aligned within the specified tolerance was a concern that was addressed via the various spacer nuts on each of the six tie rods that extend the full length of the basket. The poison material in the tube is BORAL with enriched B-10, which is not a Code material, yet integrity of the criticality control feature is required in accident scenarios. Thus, and to show long-term integrity as required by the USNRC, a fully enclosed, sealed sheathing of stainless steel encases the BORAL to ensure long-term integrity. The resulting basket design is structurally sound and criticality safe (even assuming a BORAL effectiveness factor of only 75 percent). In addition, the basket is an efficient heat transfer device and meets all of the USNRC'S requirements. The complexity of the engineering design and the analytical evaluations of the design are evident in the extensive analyses provided in the SAR and TSAR. These are some of the most-extensive licensing documents ever prepared for a spent fuel package to demonstrate compliance with 10 CFR 71 and 10 CFR 72 requirements.

A final design detail that received attention in the USNRC transport review was the copper and steel cooling fins embedded in the neutron shield. The USNRC felt strongly that the welds attaching the fins to the cask outer shell and to the neutron shield exterior skin were key to the continuous dissipation of heat and, thus, important to safety. In addition, loads on the fin weld imposed by thermal expansion of the neutron shield material between fins and loads during hypothetical accidents were evaluated to assure attachment integrity. Combined with this, the USNRC required and extensive periodic in-service inspection program designed to demonstrate continued effective heat transfer by sampling the temperature along the exterior surface of the cask.

As mentioned at the outset, the license application involved the preparation to two documents for the parallel storage and transport reviews by the USNRC and their consultants. The storage application was submitted first and thus the review got underway first. Along the way, as design modifications were incorporated to satisfy one set of USNRC requirements, both applications were required to be updated to be consistent, with an explanation for the changes in each case, and a discussion of the effect on already completed reviews. In the middle of 1992 the storage review was tentatively approved, but the transport review was not yet complete. At this same time, the previously described basket disks and tubes features were not fully included in the design of the two license applications. Additionally, as Figure IV shows, the design and licensing process is an iterative and strongly interactive one. Thus, while basket features were being finalized to meet transport requirements, all of the changes were iteratively shown to maintain or enhance design margins in each of the areas shown in the figure. The storage review was thus put on hold by the USNRC until those transport details were settled. The intent was to "catch up" the storage application when this finalization occurred. Once the transport certificate of compliance has been issued and the storage application has been updated, its conclusion and approval should follow shortly thereafter. When this occurs, it will complete the acceptability of this first-of-a-kind cask for its multifunction of storage and transport.

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R&D have amounted to some 6 million dollars annually.

- **The results**

The establishment of permanent avenues for communication within the Sector, in an attempt through UNESA to minimize the impact of diversity on NPP safety and economy, has constituted a means to counteract one of the weak points of Spain's nuclear park. At the same time, the emergence of research initiatives in the UNESA framework, close to plant operation, offsets the tending to segregate R&D activities from production, in part the cause of the main problem facing the Spanish science-technology-industry system. This approach - coordinated action in the area of operations - offers the additional and double advantage of guaranteeing that the standards established coincide with the interests of the operator and of reducing the impact of external pressures [9].

One clear output of the system has been a substantial increase in the number of projects and of the need for financing, with the situation changing from some 20 projects and 20 million dollars during the period covered by the previous R&D program to some 35 projects and 120 million dollars at present (on-going and foreseen projects).

Although this is symptomatic of progressive acceptance of R&D by the companies, it has led to a collapse of the financing system described above and to a temporary interruption of the process of approval of new projects via this route. For certain projects a new financing formula is beginning to be adopted. This method, which is covered entirely by the budgets of the NPP's themselves, consists of applying a method of financing by shares (1/3 per site and 2/3 on the basis of capacity).

Other characteristics observed have been, on the one hand, the existence of shortcomings in the final phase of the research process, for example in the marketing of the products of the research, and, on the other, long response times: the period elapsing between the identification of a need and the beginning of the corresponding project has traditionally been excessive. In theory this favors the quality of decisions, but may be the origin of lost technological opportunities (although this risk is not so obvious as in the case of other sectors, such as software or electronic consumer goods).

In concluding, a positive balance may be said to exist as regards the R&D actions taken to date, which have contributed to maintaining and enhancing the most important asset of Spanish industry: the high level of capability of its technical staff.

The excellent operating record of the Spanish nuclear power plants, the international presence of the country's industry and of the electricity utilities themselves, and the establishment of a fluid dialogue with the regulatory authority, on the basis of technically well-founded positions and decisions taken with autonomy are, to a large extent, the result of the technological stock accumulated. In this respect, the nuclear power plants and the rest of the Spanish nuclear industry do not fall into the general pattern of the country, in which technological independence is reduced. The shortcomings affecting the system, some the result of national cultural factors and others of the specific framework of the Electricity Sector, have been identified and are being counteracted via a change in mentality and organizational structures in the Sector.

(9)Among others, external forces coming from R&D contractors, due to the complexity of the technologies involved and the difficulty of obtaining economies of scale.

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