Analysis of DEC-A sequences in a NuScale-like SMR considering ATF fuel performance using the system code TRACE

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**Abstract**

This analysis assesses the benefits derived from implementing FeCrAl cladding material under DEC-A sequence conditions in a NuScale-like Small Modular Reactor, utilizing the TRACE system code, in comparison to the conventional Zr alloy fuel rod case. In a collaboration between NFQ ADVISORY SERVICES S.L. (NFQ) and the research group from the Universidad Politécnica de Madrid (UPM), an in-house version of the TRACE code is developed in which FeCrAl material properties and behavioral models code has been implemented allowing to analyse the FeCrAl cladding performance under the selected scenario for the simulations. The chosen accidental scenarios involve LOCA sequences combined with various failures in the Reactor Recirculation Valves, Reactor Vent Valves, and the Control Volume and Chemical System. The results demonstrate an increase in the available times and a general improvement in the behavior of fuel rods within the selected ATF concept, as opposed to the case where conventional Zr-alloy fuel rods are considered.

## INTRODUCTION

Given the growing interest in the Small Modular Reactors (SMR) along with the assessment of the performance of the Advanced Technology Fuels (ATFs) concepts, the simulation of a LOCA sequence in a NuScale-like SMR leading to core damage and based on the rupture of the discharge line of the Chemical Volume Control System (CVCS) and on the fail to open of the Reactor Recirculation Valves (RRVs) is addressed. By doing so and in combination with the consideration of the evolutionary ATF cladding material known as FeCrAl, it has been possible to preliminary assess the increase in the coping time caused by the presence of the commented advance material under the postulated scenario. To do so, it was needed to implement the FeCrAl material properties and behavioural models in the TRACE code by means of the modification of the source code. At this point, it should be mentioned that this work has been developed as a result of a closed collaboration between the NFQ Advisory Services S.L. company and the Universidad Politécnica de Madrid (UPM).

Additionally, the same LOCA case but considering the availability of the different configurations of RRVs have been also investigated and the results show that the opening of two out of three Reactor Vent Valves (RVVs) along with one out of two RRVs of the emergency core cooling system is enough to deal with the postulated scenario.

## nUSCALE POWER MODULE DESCRIPTION

NuScale is a GEN III+ Light Water-cooled SMR (LW-SMR) in which highly distinctive features have been introduced, such as natural circulation and fully passive safety systems. In this sense, it can be said that it is a completely disruptive design when compared to currently operating Pressurized Water Reactors (PWRs). Furthermore, NuScale remains the only design to have obtained the official approval from the U.S. NRC and is intended to operate in plants where multiple units coexist simultaneously, with a maximum of twelve units in its most powerful version [1].

The smallest independent unit in a NuScale power plant capable of producing electrical power is called the NuScale Power Module (NPM). In that sense, it should be noted that each NPM has its own CVCS responsible for controlling the boron concentration in the Reactor Coolant System (RCS), the RCS water inventory, the RCS pressure by means of the Pressurizer (PZR) spray and the amount of non-condensable gases accumulated in the PZR steam bubble during normal operation [4], steam supply system including the dedicated turbine and, the passive safety systems, except for the reactor pool, which is common to all the units forming the plant.

Regarding the physics involved in the functioning of each NPM, the idea is terribly simple and very elegant. Each NPM consists of an integral-PWR with a small core formed by 37 17x17 fuel assemblies (PWR-like but only two meters long) with a total rated power of 160 MWth [2] (currently uprated to 200 MWt) that relies on the natural circulation phenomenon to establish the mass-flow rate that cools the core, in other words, the entire RCS, including the PZR, is integrated into the Reactor Pressure Vessel (RPV) and that the RCS pumps of the current Generation II PWR designs have been removed [1]. Therefore, the gravity-driven mass-flow rate through the RCS is established by means of the density gradients and the differences in the heights at which the core and the steam generators are located [3]. To do so, the water inventory in the small core located at the bottom of the RPV is heated up and, consequently, forced to flow upwards through the central section called ‘riser’ or ‘hot leg’. Once the fluid reaches the top of the riser, it is radially redirected to the annular region formed by the outer surface of the riser structure and the inner surface of the RPV where the helically coiled steam generators (HCSGs) are located (the HCSGs are mounted surrounding the riser and the tube bundles of the two SGs are intertwined avoiding any potential asymmetrical effects in the flow path within the RCS [3]). By the action of the HCSGs, the fluid in the RCS side (outer region of the HCSGs tubes) is efficiently cooled becoming denser and flowing downwards through the downcomer region reaching the lower plenum where it is redirected to the core inlet plate and the natural circulation loop results eventually closed, further details can be found in [3]. Finally, the flow path within the RCS can be seen in *FIG. 1*.

 

*FIG. 1. General arrangement of a NPM including the CNV vessel (figure modified from [1])*

Additionally, the reader should keep in mind that the RPV is located within the interior volume of a steel Containment Vessel (CNV) partially submerged in the reactor pool which is designed to remove the heat from the core under accidental conditions acting as the ultimate heat sink [5], see *FIG. 1*. Furthermore, CNV works under vacuum condition during the normal operation of the NPM to minimize the heat losses and it is used as an additional safety system able to provide an enhanced protection against overpressure transients and a heat transfer path between the RPV and the reactor pool [5].

With regard to the passive safety systems, two dedicated systems can be found at each NPM, the Decay Heat Removal System (DHRS) and the Emergency Core Cooling System (ECCS): On one hand, the DHRS is devoted to removing the decay heat generated when the reactor is tripped along with the unavailability of the normal core cooling system through the secondary side, see [3]. DHRS is composed by two independent trains with a heat exchanger each connected to the steam and feedwater lines, respectively, mounted in the outer surface of the CNV and immersed within the reactor pool, see *FIG. 2*. During normal operation, the DHRS system is isolated from the lines of the secondary side by means of the DHRS actuation valves. When the DHRS actuation is demanded, the Main Steam and FeedWater Isolation Valves (MSIVs and FWIVs) should close and the DHRS actuation valves open allowing a natural circulation mass-flow rate to be established for the steam leaving the top of the HCSGs and driven by the density gradient (caused by the cooling capability of the DHRS heat exchangers) and the difference in height between the bottom of DHRS heat exchangers and the inlet of the SGs. Finally, the condensate from the DHRS heat exchangers is re-injected to the feedwater lines of the HCSGs, see [3].

On the other hand, the ECCS has been specifically designed to deal with failures of the normal core cooling system and, specially, with LOCA or low temperature overpressure transients, see [5]. This system is formed by three Recirculation Vent Valves (RVVs) and two Reactor Recirculation Valves (RRVs) located at the top head of the RPV and around 1.8 meters above the Top of the Active Fuel (TAF) height in the RPV outer surface [5] respectively. An additional Inadvertent Actuation Block (IAB) feature is mounted in each ECCS valve to prevent its potential spurious opening under normal operating conditions [5]. In accordance with [6], the success criteria of the ECCS system consists of the opening of two out of the three RVVs and one out of the two RRVs. Once the ECCS valves open, the depressurization of the RCS is achieved by means of the discharge of part of the RCS inventory into the CNV volume where it is condensed by the contact with the CNV walls and collected at lower CNV region. For that reason, the liquid level in the CNV is increased and when the proper height is reached, a mass-flow rate from the CNV to the RPV and, more in particular, to the downcomer ring through the RRVs is established allowing to refill the core and riser regions, see *FIG. 2*.

|  |  |
| --- | --- |
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*FIG. 2. ECCS (left) and DHRS (right) performances [1]*

## DESCRIPTION OF THE LOCA SECUENCES IN NUSCALE

In a NuScale-like SMR, the Large-Break LOCA is fully avoided by the integral design of the RPV and the most relevant are the ones involving ruptures of lines inside the CNV since they cannot be isolated through the actuation of the module protection system. In that sense, there are only pipes with small diameters within the RCS going through the CNV such as the CVCS makeup and letdown lines, the RPV high point vent line, or the pressurizer spray supply line [7]. For those reasons, in addition to the break size, the LOCA accidents in NuScale can be classified in terms of the fluid condition, steam or liquid, in direct contact with the rupture.

The evolution and consequences of the LOCA accidents in a NuScale-like SMR are similar, only slight differences depending on the location of the break are found, more in particular, if it took place in a steam space such as in the case of the RPV high point vent or the pressurizer spray supply lines, the ECCS actuation is commonly demanded by ‘low PZR pressure’ signal. On the other hand, if a liquid space is in direct contact with the break, as in the case of the rupture of the CVCS makeup or discharge lines, the most likely ECCS signal to be triggered the first is ‘High CNV level’, further details can be found in [7].

A LOCA accident in NuScale-like SMR starts with the rupture of one of the previously commented lines causing a discharge of the RCS inventory into the CNV. For that reason, the pressure along with the liquid level within the CNV internal volume is increased and the ECCS actuation is demanded. Once the IAB differential pressure threshold is reached, the ECCS valves will open allowing the pressure within the RPV and CNV to be equalized. As the RRVs are now open, the inventory within the CNV can be re-injected in the RCS and a natural circulation loop is established between the RPV and the CNV driven by the decay heat produced within the core and the cooling capability of the CNV walls in contact with the reactor pool. In addition, the DHRS actuation is not credited in the analysis of this kind of accidents in a NuScale-like SMR demonstrating that the post-LOCA long-term core cooling can be completely accomplished by the actuation of only the ECCS performance.

## MODELLING OF the NUSCALE POWER MODULE USING TRACE

A full-plant model of a NuScale-like SMR has been built for the system code TRACE using a combination of 1D and 3D hydraulic components and HEAT STRUCTURE components to model the heat transfer and the thermomechanical features of the fuel rods with Zr-4 and FeCrAl claddings. In this model, the RCS is simulated by means of three 3D VESSEL components, two for the RCS regions and one for the core, with the following features, see *FIG. 3*:

* The core region is modelled using a 3D Cartesian VESSEL with 20 axial levels and a square matrix of 7x7 allowing to achieve a 1:1 ratio between the thermal-hydraulic channels in the VESSEL and the number of fuel assemblies. In that sense, each fuel assembly has been implemented in the model using the corresponding HEAT STRUCTURE components in which the thermomechanical parameters related to the pre-transient fuel rod geometry, internal pressure and materials are implemented. Three cases are simulated: UO2/Zr-4; UO2/FeCrAl (same geometry) and UO2/FeCrAl with a reduced cladding thickness (to compensate the impact of the higher cross-section of neutron capture of FeCrAl). To perform the FeCrAl calculations, the material properties found in the literature were implemented in the TRACE source code [9-14];
* The respective data regarding the fuel rod geometry of the different cases simulated can be seen in TABLE 1;

TABLE 1. Fuel rod data for the different cases.

|  |  |  |
| --- | --- | --- |
| Parameter (SI units) | UO2/Zr-4|FeCrAl | UO2/FeCrAl(Reduced thickness) |
| Fuel pellet Outer Diameter (mm) | 8.115 | 8.115 |
| Cladding Inner Diameter (mm) | 8.280 | 8.280 |
| Cladding Outer Diameter (mm) | 9.499 | 9.05 |
| Cladding thickness (mm) | 0.610 | 0.385 |
| Fill Gas | He | He |
| Fill Gas Pressure (MPa) | 1.482 | 1.482 |

* Upper RPV region includes the regions associated with the upper riser, the annular region with the primary side of the HCSGs and the upper downcomer section. It is simulated using the ‘Tube Bank Crossflow’ 3D cylindrical VESSEL component and it is built with 45 axial levels, 2 radial rings (1 for the downcomer and 1 for the riser region) and 4 azimuthal sectors;
* Lower RPV region simulates the annular region with the lower part of the downcomer, the lower plenum, the core and the lower and transition riser regions. This VESSEL is built with 18 axial levels, 2 radial rings (1 for the downcomer and 1 for the core) and 4 azimuthal sectors;
* The PZR is implemented in the TRACE model using a 1D ‘Pressurizer’ kind of PIPE component.

As it can be seen in *FIG. 3*, the secondary side is explicitly modelled up to the turbine stop valve. The interior of the HCSGs has been modelled using the ‘curved pipe’ PIPE components to model the special geometry of those tubes and, the CNV is simulated using a 3D cylindrical VESSEL with 24 axial levels, 3 radial rings and 1 azimuthal sector to allow the flow patterns after the rupture to be developed. Finally, the piping and heat exchangers of the DHRS have been explicitly modelled along with the reactor pool using 1D PIPEs components.



*FIG. 3. Nodalization scheme of the full-plant TRACE model*

### Steady-State calculation

A null-transient calculation of 6000 seconds is simulated considering the following assumptions:

* The initial status of the reactor is Hot Full Power (100% of the nominal power) at the Beginning of Cycle (BOC) condition determining the radial and axial power profiles, see [2];
* Nominal values are assumed for the plant parameters except for the RCS mass flow rate which is restrained to its minimum value (535.24 kg/s);
* The CVCS letdown mass flow rate is assumed to be equal to the CVCS makeup mass flow rate which is restrained to its maximum value (3.15 kg/s);
* The CNV pressure under normal operation condition remains at 0.21 bar.

A summarizing table with the comparison of the results and the reference values is included below.

TABLE 1. Comparison of steady-state calculation parameters with the reference values from the DCA.

|  |  |  |
| --- | --- | --- |
| Parameter (SI units) | DCA | TRACE (Error %) |
| RCS pressure (MPa) | 12.755 | 12.732 (-0.18) |
| RCS mass-flow rate (kg/s) | 535.24 | 536.98 (0.32) |
| Core mass-flow rate (kg/s) | 496.17 | 501.52 (1.08) |
| Core inlet temperature | 531.48 | 531.59 (0.02) |
| Steam temperature (K) | 580.04 | 585.45 (0.93) |
| SG pressure (MPa) | 3.447 | 3.443 (-0.11) |

## modelling of a LOCA SEQUENCE in the cvcs discharge line of A NUSCALE-LIKE SMR

In this work, the scenario to be simulated consists in a LOCA sequence caused by the rupture of the CVCS discharge line within the CNV in which the opening of two out of three RVVs is accomplished, however, the failure of the opening of both RRVs is assumed. The LOCA transient begins 50 s after a steady-state calculation of 6000 s with the discharge of the RCS liquid inventory into the CNV due to the break of the CVCS discharge line. For that reason, the collapsed water level along with the pressure is rapidly increased, as it is depicted in *FIG. 4*. After of approximately another 50 s of simulation the reactor is tripped because the ‘High CNV Pressure’ setpoint (~0.65 bar) is reached and, 400 seconds later, the ‘High CNV Level’ signal is triggered and the ECCS actuation is demanded and when the IAB pressure threshold is below 62 bars, the ECCS valves are allowed to open. As the opening of the RRVs is not considered in the base case, only the opening of two RVVs is achieved and, for that reason, the pressure within the RPV and CNV are equilibrated. Finally, as there is no possibility to re-inject the liquid water in the CNV to the RPV by means of the RRVs, the core is eventually uncovered and the cladding temperature is increased up to reaching the acceptance criterion of 1477 K around 5450 s after the beginning of the LOCA transient leading to a core damage condition, see *FIG. 5*.

A very similar performance regarding the evolution of the CNV level and cladding temperatures have been obtained in the cases in which one/two RVVs are available. According to the results, the opening of two RVVs and one RRV is enough to cope with the postulated LOCA scenario. On the other hand, the increase in the coping time due to the presence of the FeCrAl cladding is demonstrated in the results, as it can be seen in *FIG. 5*. In fact, the case of UO2/FeCrAl cladding with the reduced thickness generates a delay in the time to reach the core damage condition of ~8 min while in the case of UO2/FeCrAl in which the same geometry as in the UO2/Zr-4 case is maintained, an extension in the coping time of ~11 min is obtained.

|  |  |
| --- | --- |
| High CNV Level Signal (ECCS) | High CNV Pressure (SCRAM)Low wide range pressure signal (ECCS)Low PZR Pressure signal (SCRAM) |

*FIG. 4. Collapsed water level within the CNV (left) and system/CNV pressure evolution (right)*

|  |  |
| --- | --- |
| Acceptance criterion (PCT = 1477 K) | Gráfico  Descripción generada automáticamente con confianza bajaAcceptance criterion (PCT = 1477 K) |

*FIG. 5. Peak Cladding Temperature (PCT) varying the number of RRVs available (left) and cladding materials (right)*

## Conclusions

The main conclusions drawn from this preliminary analysis are listed below:

* In this study, a LOCA accident in a NuScale-like SMR based on the rupture of the CVCS discharge line has been simulated with a full-plant TRACE model in which the fail to open of the RRVs leads directly to core damage after around 5500 s;
* The simulation of the cases in which Zr-4 and FeCrAl materials are considered for the cladding have demonstrated a slight improvement in the coping time due to the presence of FeCrAl. In that sense, an improvement of only several minutes is achieved;
* Two additional cases have been simulated considering the opening of the different configurations of the RRVs and, it has been demonstrated that with only one RRV available, the NuScale-like SMR is able to deal with the postulated scenario conditions.

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