NUREG/IA-0482

International Agreement Report

Using TRACE, MELCOR, CFD, and FRAPTRAN to Establish the Analysis Methodology for Chinshan Nuclear Power Plant Spent Fuel Pool

Prepared by:
Jong-Rong Wang, Chunkuan Shih, Shao-Wen Chen, Yung-Shin Tseng, Yu Chiang, Wan-Yun Li, Jung-Hua Yang, Hsiung-Chih Chen, Show-Chyuan Chiang*, Tzu-Yao Yu*

Nuclear and New Energy Education and Research Foundation; Institute of Nuclear Engineering and Science, National Tsing Hua University
101 Section 2, Kuang Fu Rd., HsinChu, Taiwan

*Department of Nuclear Safety, Taiwan Power Company
242, Section 3, Roosevelt Rd., Zhongzheng District, Taipei, Taiwan

K. Tien, NRC Project Manager

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Manuscript Completed: April 2017
Date Published: October 2018

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Library at www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and Title 10, "Energy," in the *Code of Federal Regulations* may also be purchased from one of these two sources.

1. **The Superintendent of Documents**
   U.S. Government Publishing Office
   Washington, DC 20402-0001
   Internet: bookstore.gpo.gov
   Telephone: (202) 512-1800
   Fax: (202) 512-2104

2. **The National Technical Information Service**
   5301 Shawnee Road
   Alexandria, VA 22312-0002
   www.ntis.gov
   1-800-553-6847 or, locally, (703) 605-6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: **U.S. Nuclear Regulatory Commission**
Office of Administration
Multimedia, Graphics, and Storage &
Distribution Branch
Washington, DC 20555-0001
E-mail: distribution.resource@nrc.gov
Facsimile: (301) 415-2289

Some publications in the NUREG series that are posted at NRC's Web site address www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852-2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036-8002
www.ansi.org
(212) 642-4900

NUREG/IA-0482

International Agreement Report

Using TRACE, MELCOR, CFD, and FRAPTRAN to Establish the Analysis Methodology for Chinshan Nuclear Power Plant Spent Fuel Pool

Prepared by:
Jong-Rong Wang, Chunkuan Shih, Shao-Wen Chen, Yung-Shin Tseng, Yu Chiang, Wan-Yun Li, Jung-Hua Yang, Hsiung-Chih Chen, Show-Chyuan Chiang*, Tzu-Yao Yu*

Nuclear and New Energy Education and Research Foundation; Institute of Nuclear Engineering and Science, National Tsing Hua University
101 Section 2, Kuang Fu Rd., HsinChu, Taiwan

*Department of Nuclear Safety, Taiwan Power Company
242, Section 3, Roosevelt Rd., Zhongzheng District, Taipei, Taiwan

K. Tien, NRC Project Manager

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Manuscript Completed: April 2017
Date Published: October 2018

Prepared as part of
The Agreement on Research Participation and Technical Exchange
Under the Thermal-Hydraulic Code Applications and Maintenance Program (CAMP)

Published by
U.S. Nuclear Regulatory Commission

ABSTRACT

Chinshan nuclear power plant (NPP) is the first plant in Taiwan and is a BWR/4 plant. People have more concern for the NPPs safety in Taiwan after Fukushima NPP disaster happened. Hence, we established the analysis methodology and performed the safety analysis of Chinshan NPP SFP (spent fuel pool) by using TRACE, MELCOR, CFD, and FRAPTRAN codes. There were two steps in this study. The first step was the establishment of Chinshan NPP SFP models by using TRACE, MELCOR, and CFD. Then, under the SFP cooling system failure condition (Fukushima-like accident), the transient analysis was performed. Additionally, the sensitive study of the time point for water spray was also performed. The next step was the analysis of the fuel rod performance by using FRAPTRAN and TRACE analysis results. Finally, the animation model of Chinshan NPP SFP was presented by using the SNAP animation function with MELCOR's results.

FOREWORD

The US NRC is developing an advanced thermal hydraulic code named TRACE for nuclear power plant safety analysis. The development of TRACE is based on TRAC, integrating RELAP5 and other programs. NRC has determined that in the future, TRACE will be the main code used in thermal hydraulic safety analysis, and no further development of other thermal hydraulic codes such as RELAP5 and TRAC will be continued. A graphic user interface program, SNAP which processes inputs and outputs for TRACE is also under development. One of the features of TRACE is its capacity to model the reactor vessel with 3-D geometry. It can support a more accurate and detailed safety analysis of nuclear power plants. TRACE has a greater simulation capability than the other old codes, especially for events like LOCA.

Taiwan and the United States have signed an agreement on CAMP (Code Applications and Maintenance Program) which includes the development and maintenance of TRACE. To meet this responsibility, the models of Chinshan NPP SFP were built by using TRACE, MELCOR, CFD, and FRAPTRAN codes. In this report, these models were used to evaluate the SFP cooling system failure transient (Fukushima-like accident).

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

EXECUTIVE SUMMARY

An agreement in 2004 which includes the development and maintenance of TRACE was signed between Taiwan and USA on CAMP. NTHU is the organization in Taiwan responsible for applying TRACE to thermal hydraulic safety analysis in order to provide users' experiences and development suggestions. To fulfill this responsibility, the TRACE, MELCOR, CFD, and FRAPTRAN models of Chinshan NPP SFP were established.

According to the manuals [1]-[2], TRACE is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. The 3-D geometry model of reactor vessel, which is one of the representative features of TRACE, can support a more accurate and detailed safety analysis of NPPs.

Sandia National Laboratories developed MELCOR [3]. MELCOR can model the progression of NPPs severe accidents. MELCOR can treat the broad spectrum of severe accident phenomena in NPPs. FRAPTRAN can calculate the performance of fuel rods during the transients and hypothetical accidents such as anticipated transients without scram (ATWS), LOCA, and reactivity-initiated accidents (RIAs) [4].

Additionally, SNAP is a graphic user interface program and can process the inputs, outputs, and animation models for TRACE, MELCOR and FRAPTRAN. Therefore, Chinshan NPP SFP models were built by TRACE, MELCOR, and FRAPTRAN with SNAP in this study.

Chinshan NPP is the first plant in Taiwan and is a BWR/4 plant. The original rated power of Chinshan NPP for each unit is 1775 MWt. Chinshan NPP finished the SPU (stretch power uprate) project and the operating power is 1840 MWt now. The SFP cooling system failed and the SFP safety issue generated in the Fukushima NPP disaster. Hence, there were two steps in this study. The first step was the establishment of Chinshan NPP SFP models by using TRACE, MELCOR, and CFD. Then, under the SFP cooling system failure condition (Fukushima-like accident), the transient analysis was performed. The sensitive study of the time point for water spray was also performed. The next step was the analysis of the fuel rod performance by using FRAPTRAN and TRACE analysis results. Finally, the animation model of Chinshan NPP SFP was presented by using the SNAP animation function with MELCOR's results.

ABBREVIATIONS

ATWS	Anticipated Transients Without Scram
BAF	Bottom of Active Fuel
CAMP	Code Applications and Maintenance Program
LOCA	Loss Of Coolant Accidents
NPP	Nuclear Power Plant
NTHU	National Tsing Hua University
RIA	Reactivity-Initiated Accidents
SFP	Spent Fuel Pool
SPU	Stretch Power Uprate
TAF	Top of Active Fuel
URG	Ultimate Response Guideline

1 INTRODUCTION

U.S. NRC developed TRACE code for NPP thermal hydraulic analysis [1]-[2]. According to the TRACE's manuals [1]-[2], TRACE is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. The 3-D geometry model of reactor vessel, which is one of the representative features of TRACE, can support a more accurate and detailed safety analysis of NPPs. TRACE also provides greater simulation capability than the previous codes, especially for events like LOCA. Additionally, TRACE was used to simulate the SFP of Fukushima NPP according to Sandia National Laboratories report [5].

Sandia National Laboratories developed MELCOR [3]. MELCOR can model the progression of NPPs severe accidents. MELCOR can treat the broad spectrum of severe accident phenomena in NPPs. These include thermal-hydraulic responses in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior. According to the reference [6], MELCOR was used to perform the study of SFP of Fukushima Dai-ichi Unit 4. The reference [7] shows that MELCOR was used to establish the SFP model of a Nordic BWR. The loss-of-pool-cooling accidents was simulated and analyzed by this model. The above studies indicate that MELCOR is capable of handling the simulation of the SFP.

FRAPTRAN can calculate the performance of fuel rods during the transients and hypothetical accidents such as ATWS, LOCA, and RIAs [4]. Additionally, SNAP is a graphic user interface program and can process the inputs, outputs, and animation models for TRACE, MELCOR and FRAPTRAN. Therefore, Chinshan NPP SFP models were built by TRACE, MELCOR, and FRAPTRAN with SNAP in this study.

Chinshan BWR/4 NPP was building in 1970. The original rated power of Chinshan NPP for each unit is 1775 MWt. Chinshan NPP finished the project of SPU and the operating power is 1840 MWt now. After the Fukushima NPP disaster, there is more concern for Taiwan NPPs safety. The SFP cooling system failed and the SFP safety issue generated in the Fukushima NPP disaster. Hence, we established the safety analysis methodology of Chinshan NPP SFP using TRACE, MELCOR, CFD and FRAPTRAN codes to concern the SFP safety.

There were two steps in this study. The first step was the establishment of Chinshan NPP SFP models by using TRACE, MELCOR, and CFD. Then, under the SFP cooling system failure condition (Fukushima-like accident), the transient analysis was performed. Additionally, the sensitive study of the time point for water spray was also performed. The next step was the analysis of the fuel rod performance by using FRAPTRAN and TRACE analysis results. Finally, the animation model of Chinshan NPP SFP was presented by using the SNAP animation function with MELCOR's results.

2 METHODOLOGY

Figure 1 presents the safety analysis methodology of Chinshan NPP SFP. First, Chinshan NPP SFP data and the reports were collected [8]-[10]. The geometry of SFP was 12.17 m × 7.87 m × 11.61 m and the initial condition was 60 ℃ (water temperature) / 1.013 × 105 Pa. The total power of the fuels was roughly 8.9 MWt initially. Second, TRACE/SNAP SFP model was established to perform the thermal-hydraulic analysis. The channel component was used to simulate the fuel bundles and the 3D vessel component was used to simulate the SFP. Third, MELCOR/SNAP SFP model was built to run the severe accident analysis. The amounts of hydrogen generation and the mass variation of Zr/ZrO_2 can be calculated by using MELCOR. Next, in order to estimate the thermal-hydraulic phenomenon of the local region of SFP in detail, the CFD SFP model was used to do this analysis. Finally, by using TRACE, or MELCOP, or CFD results (ex: power and coolant conditions), the fuel rod of FRAPTRAN model was established. TRACE's analysis results were used in FRAPTRAN's input files in this report.

Figure 2 illustrates the MELCOR/SNAP SFP model. This model includes 10 control volume components, one core component, and 13 heat structure components. The core component was used to model the material of racks and fuel assembles. The water of the SFP was modeled using the control volume components (CVH package). The core component was divided into 10 axial levels and 4 radial rings (see in Figure 3). The fuels were divided into 8 axial nodes which were in the level 3~10 of core component and No. 2, 102, 202 CVH components. Total 3076 fuel bundles were in the SFP. Figure 4 shows the relationship between the ring 1~3 of MELCOR core component and the fuel positions of spent fuel pool. Table 1 lists the power fraction and fuel bundles number of core component ring 1~3. The hottest fuel rod was in ring 1.

Figure 5 depicts TRACE SFP model. The 3-D vessel component of TRACE was used to simulate the pool. This 3-D vessel component included 14 axial levels, 1 X axis, and 1 Y axis. In the axial direction, the water was in the axial level 1 to level 11 and the air was in the axial level 12 to level 14. Six channel components were used to simulate the fuel bundles in this study. The channel component is a 1-D component and can simulate full length fuel rods, partial length fuel rods and water rods. The channel component was divided into 25 axial nodes. Figure 6 shows the relationship between the channel component and the fuel positions of spent fuel pool. Table 2 lists the power fraction and fuel bundles number of channel components. The hottest fuel rod was in channel 51. The heat source of the spent fuel pool was the decay heat of the fuels and was simulated by a power component. The power component used the power table to simulate the power varying during the transient. This model also had the simulation of the heat conduction between the racks of the fuels and the pool. One heat structure component of TRACE was used to simulate the heat exchange from SFP to the fuels' racks.

CFD (Fluent) SFP model was built in our previous study [8]. We depicted briefly CFD model in this paper. Figure 7 presents the 2D, 3D fuel bundle and 3D SFP model. The 2D fuel bundle model was utilized to calculate the effective thermal conductivity properties for the porous media in the 3D SFP model. Moreover, the realistic fuel arrangement was considered in the 3D spent fuel model, which provided a more reliable boundary to find the location of the hottest fuel.

We used TRACE's results (ex: power and coolant conditions data) and fuel rod geometry data to establish FRAPTRAN model in this study. Figure 8 illustrates the FRAPTRAN fuel rod model. Total 23 nodes from bottom to top were in this model. Subsequently, the fuel rod analysis was performed by FRAPTRAN. Then, we checked the cladding surface / fuel centerline temperature of TRACE and FRAPTRAN to avoid the inconsistency in the temperature trend of TRACE and FRAPTRAN. Finally, the analysis results were obtained from FRAPTRAN output file.

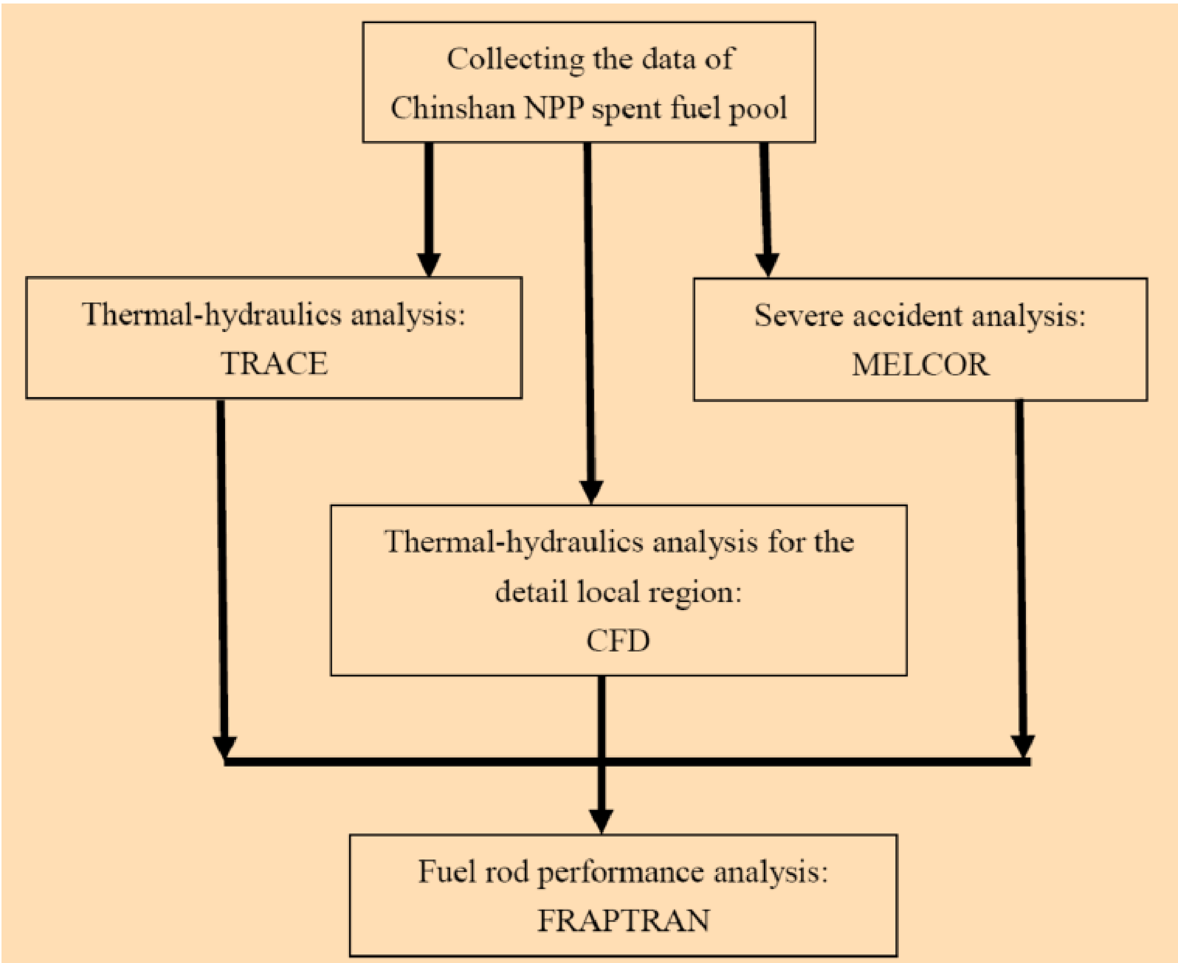


Figure 1 The Flow Chart of Analysis Methodology

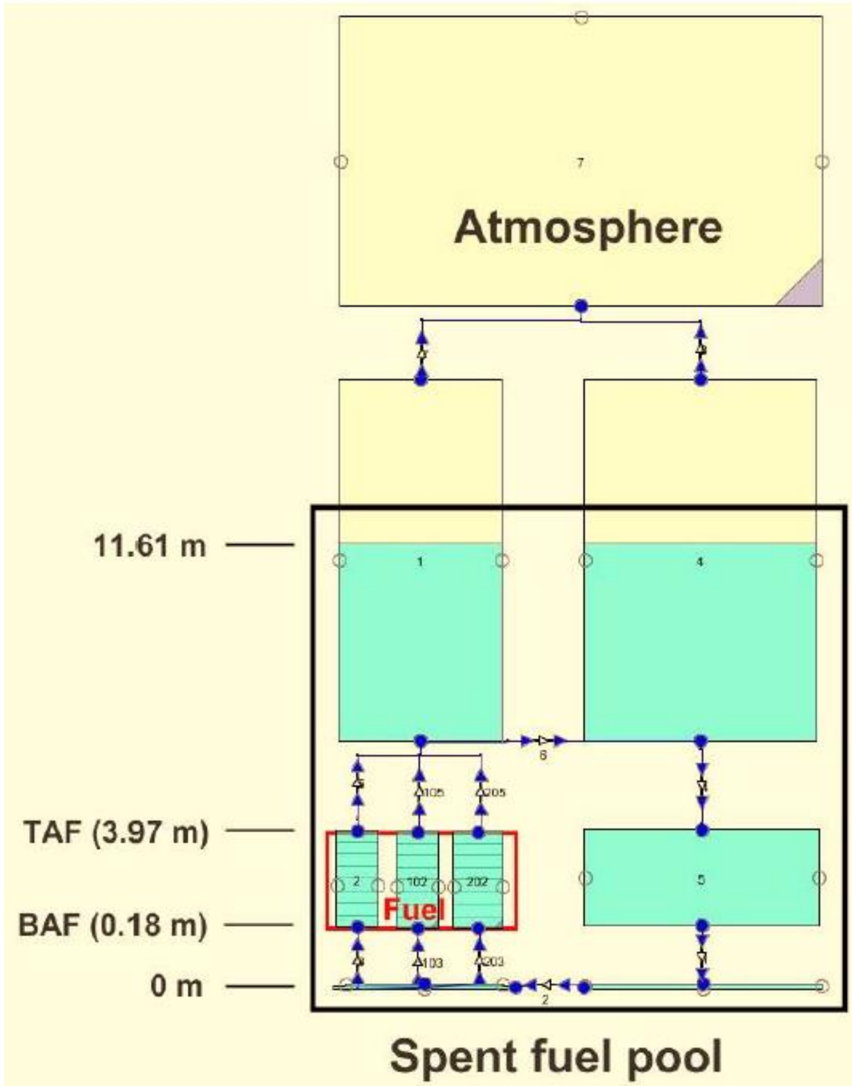
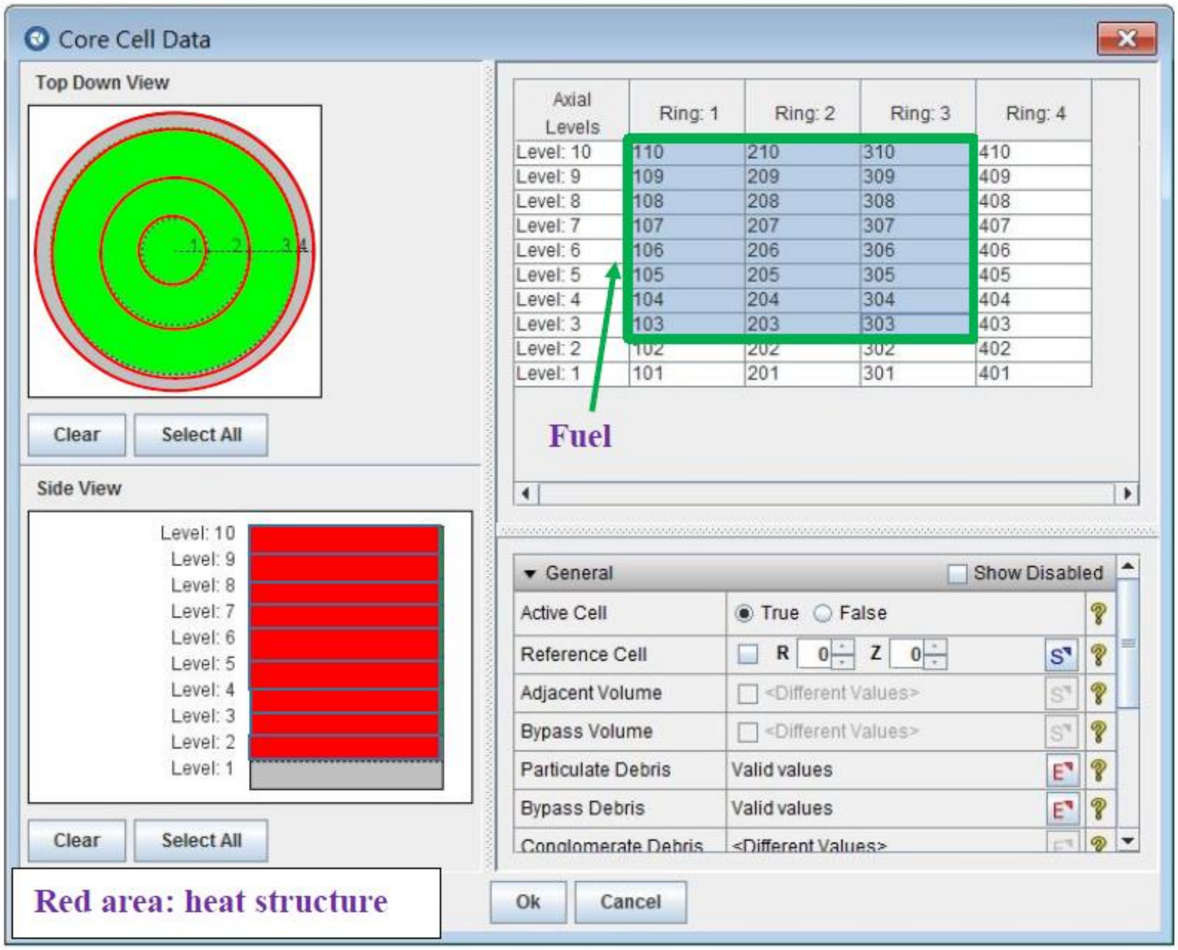
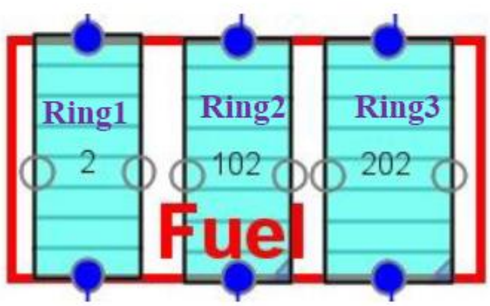


Figure 2 SFP MELCOR/SNAP Model



Core component



CVH component

Figure 3 SFP Core and CVH Components

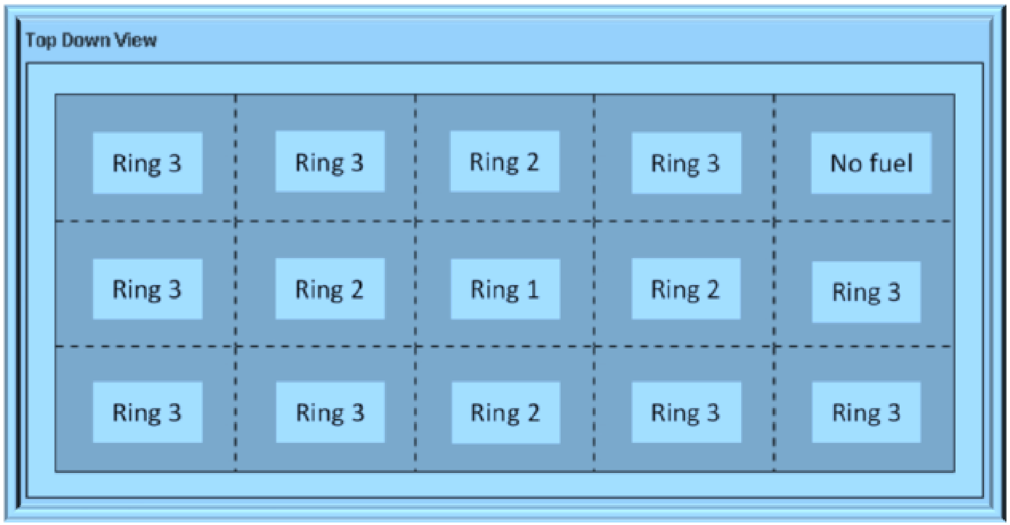


Figure 4 The Positions of Ring 1~3 in the SFP

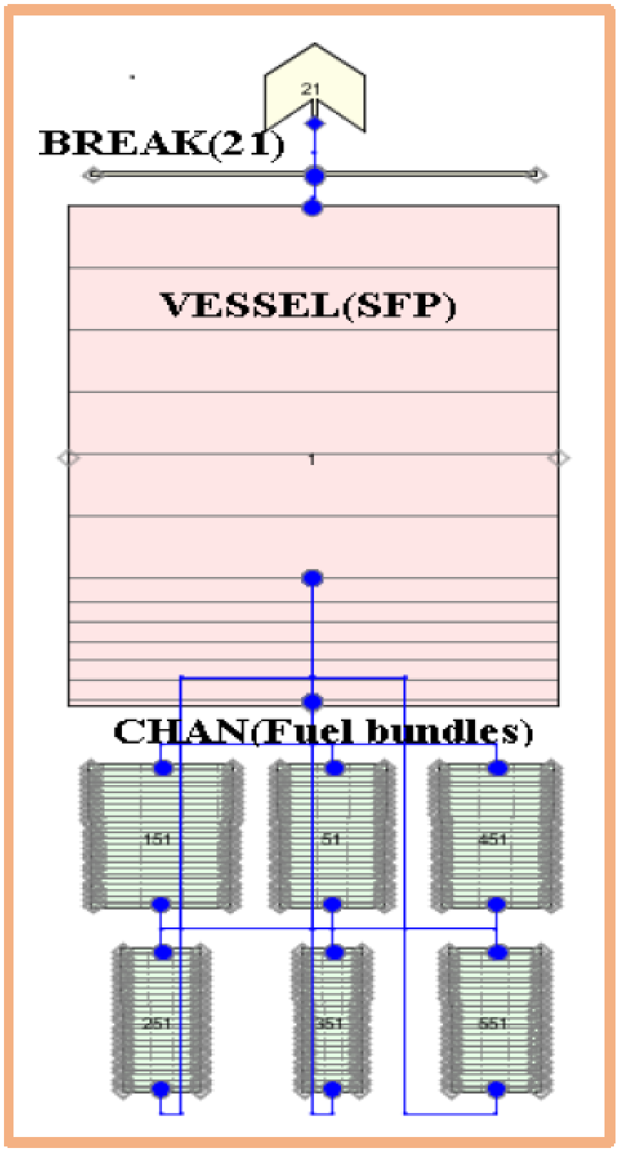


Figure 5 SFP TRACE/SNAP Model

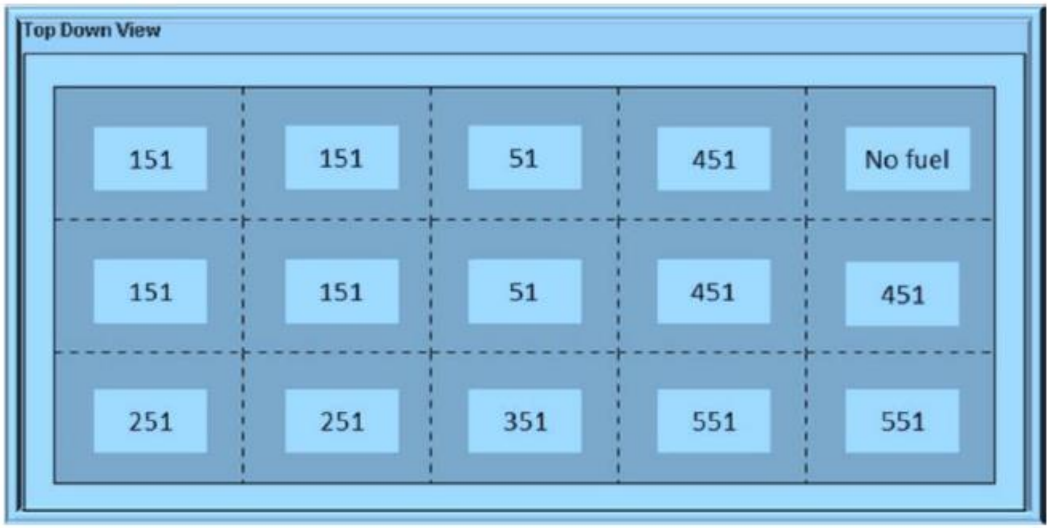


Figure 6 **The Positions of Channel Components in the SFP**

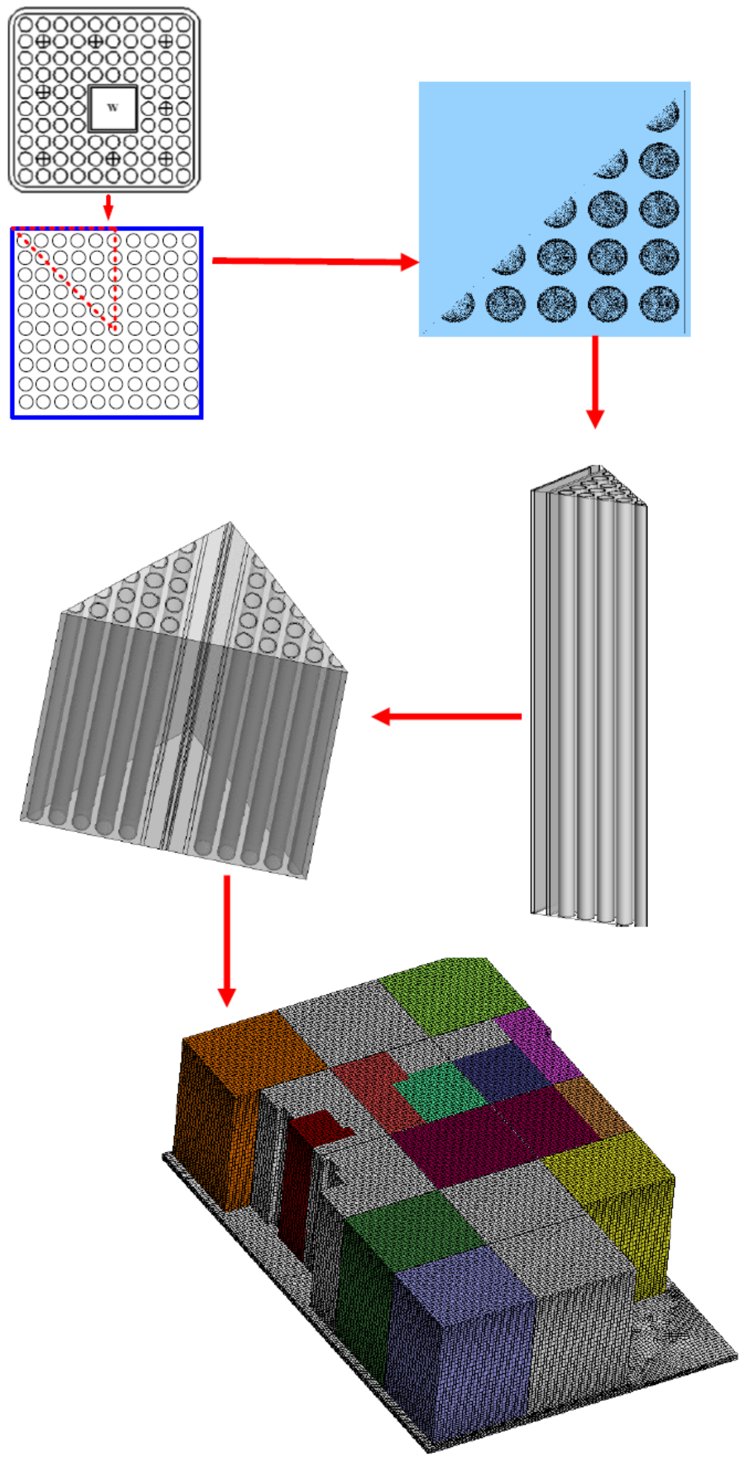


Figure 7 SFP CFD Model

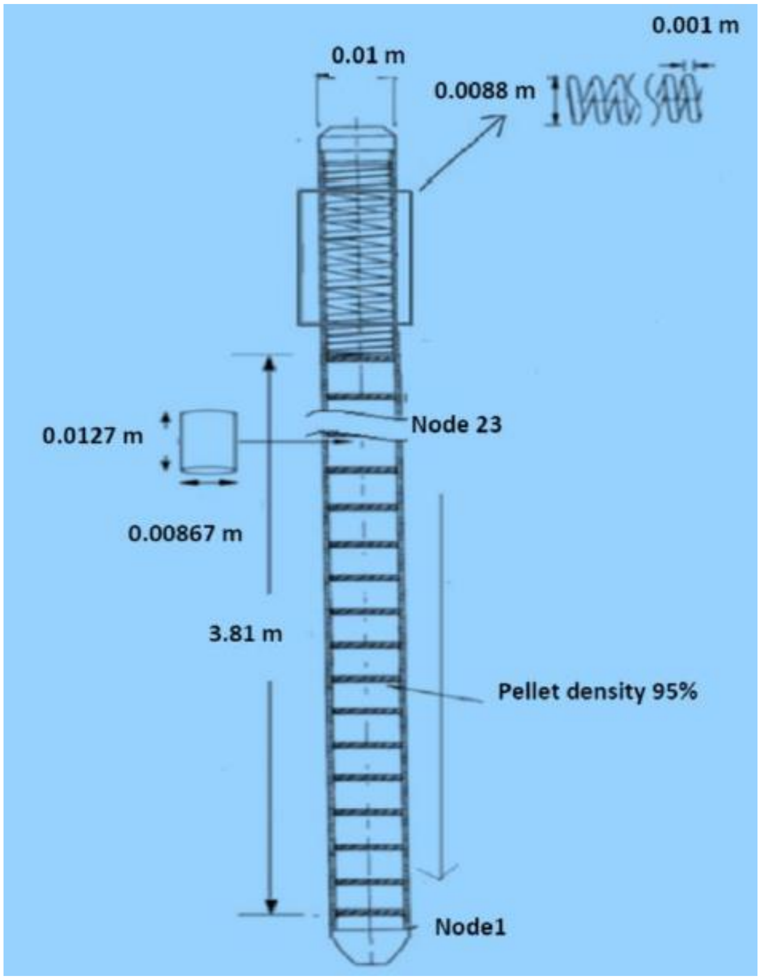


Figure 8 FRAPTRAN Fuel Rod Model

Table 1 The Power Fraction and Fuel Bundles Number for MELCOR Area

MELCOR area	Power fraction (%)	Fuel bundles number
Ring 1	49.67	353
Ring 2	34.35	862
Ring 3	15.98	1861

Table 2 The Power Fraction and Fuel Bundles Number for Channel Components

Channel	Power fraction (%)	Fuel bundles number
151	4.88	988
51	80.9	563
451	1.92	620
251	10.18	310
351	0.66	179
551	1.44	416

3 RESULTS

The SFP heat source was the fuels decay heat. Figure 9 depicts the total power of the fuels. First, the transient analysis under the SFP cooling system failure condition (Fukushima-like accident) was performed. We assumed that the SFP cooling system failed, so no water added into SFP during the transient. The heat of the fuels was removed by the evaporation of pool water. The safety issue of the fuel rods cladding may generate after the uncovered of the fuels occurred.

Figure 10 illustrates the max cladding temperature result of TRACE. The initial water temperature of SFP was 60 ℃. After the cooling system failed, the time of the cladding temperature which reached 100 ℃ was roughly 2.9 hours. Subsequently, the water dried out. This caused the water level lower than the TAF (top of active fuel). Figure 11 presents the water level results of TRACE. The water level was lower than TAF at 2.7 days. The uncovered of the fuels caused to the cladding temperature increase roughly at 3 days. Finally, the max cladding temperature reached 1088.7 K at 3.6 days. According to the URG [11], the max cladding temperature should be lower than 1088.7 K. When the max cladding temperature reached 1088.7 K, it indicates that the zirconium-water reaction of the fuels occurs. The zirconium-water reaction may make the cladding temperature increase sharply and may generate the burst of the fuel rods cladding. The above phenomenon may cause the fuels safety issue.

Figure 10 and 11 also presents the CFD and MELCOR max cladding temperature and water level results for this case. The trends of MELCOR, TRACE and CFD were similar. MELCOR's cladding temperature reached the limit: 1088.7 K at 3.7 days and CFD's cladding temperature reached the limit: 1088.7 K at 3.4 days. The above results indicated that the zirconium-water reaction was able to generate. CFD model presented a highest temperature rising trend and caused the zirconium-water reaction generation earlier than those in MELCOR and TRACE. The difference was that a higher local heat source caused an obvious temperature increase since the realistic discharge fuel arrangement had been considered in CFD model. Since a realistic fuel discharge arrangement had been considered in the simulation, the region with new discharge fuel had a higher decay heat. According to the CFD data [8], the new discharge fuels caused two obvious hot spots in central and north-west side of pool. The non-uniform temperature reasonably induced a higher temperature and faster temperature rising trend. Additionally, the differences of the above results were also caused by the different the calculation procedures, phenomenological modeling, and nodalization.

In addition, the sensitive study of the time point for water spray was also performed by using the TRACE/SNAP model. The flow rate of the water spray is 200 gpm which refers to NEI 06-12 [9]. Figure 12 and 13 illustrates the TRACE results. When the water level is lower than TAF, the cladding temperature starts to go up. However, if the water spray is performed, the cladding temperature can be kept under 1088.7 K before the water level is lower than 2/3 TAF.

Figure 14~18 present MELCOR results about zirconium-water reaction. When the zirconium-water reaction occurred, the mass of Zr decreased and ZrO_2 increased (see Figure 14 and 15). In addition, the mass of H_2 also went up after 3.7 days (see Figure 16). Total amounts of hydrogen generation were about 570 kg (4 days) in this case. According to the reference [5], MELCOR results depicted that total amounts of hydrogen generated was about 2000 kg in SFP of Fukushima Dai-ichi Unit 4. The reference [5] also indicated that enough hydrogen (150 kg) was able to produce an explosion. In our case, enough hydrogen (150 kg) was generated at

3.82 days. Finally, because the zirconium-water reaction occurred, the larger heat amount was observed after 3.7 days (see Figure 17 and 18).

Figure 19~23 show the FRAPTRAN results. By using TRACE analysis results as the input data of FRAPTRAN, FRAPTRAN can calculate the fuel rod performance in detail. In FRAPTRAN analysis, Case 1 is the water spray at TAF; Case 2 is the water spray at 2/3TAF; Case 3 is no water spray. According to FRAPTRAN analysis results, the highest cladding temperature located on the node 21. FRAPTRAN out files also present that the cladding burst roughly at 3.7 days for Case 3. Figure 19 depicts that oxide thickness increased after 3.7 days for Case 3. The oxide thickness increased due to the zirconium-water reaction occurred (see Figure 20). The cladding temperature affected the hoop strain and stress of cladding and the structural radial gap. Figure 21 and 22 show the cladding hoop strain and stress results. When the cladding temperature increased, the cladding hoop strain and stress also increased. NUREG-0800 Standard Review Plan [10] clearly defines fuel cladding failure criteria. For the uniform strain value, it is limited not to exceed 1%. The cladding hoop strain went up sharply after 3.5 days and was larger than this limit. The cladding hoop stress drops abruptly to zero after 3.7 days. Figure 23 depicts the structural radial gap results. The structural radial gap became 3 mm at about 3.7 days. It indicated that cladding rupture generated. The above results also implied that the integrity of cladding did not keep after 3.7 days.

Figure 24 presents the MELCOR/SNAP animation model. This animation model can present the MELCOR results which include water level, cladding temperature, ZrO_2 and H_2 mass. Figure 24 (a) depicts the transient beginning. The water level decreased after the transient started. The increase in the cladding temperature, ZrO_2 and H_2 mass were observed in Figure 24 (b) when the water level was lower than TAF. Finally, the integrity of cladding did not keep (see Figure 24 (c)).

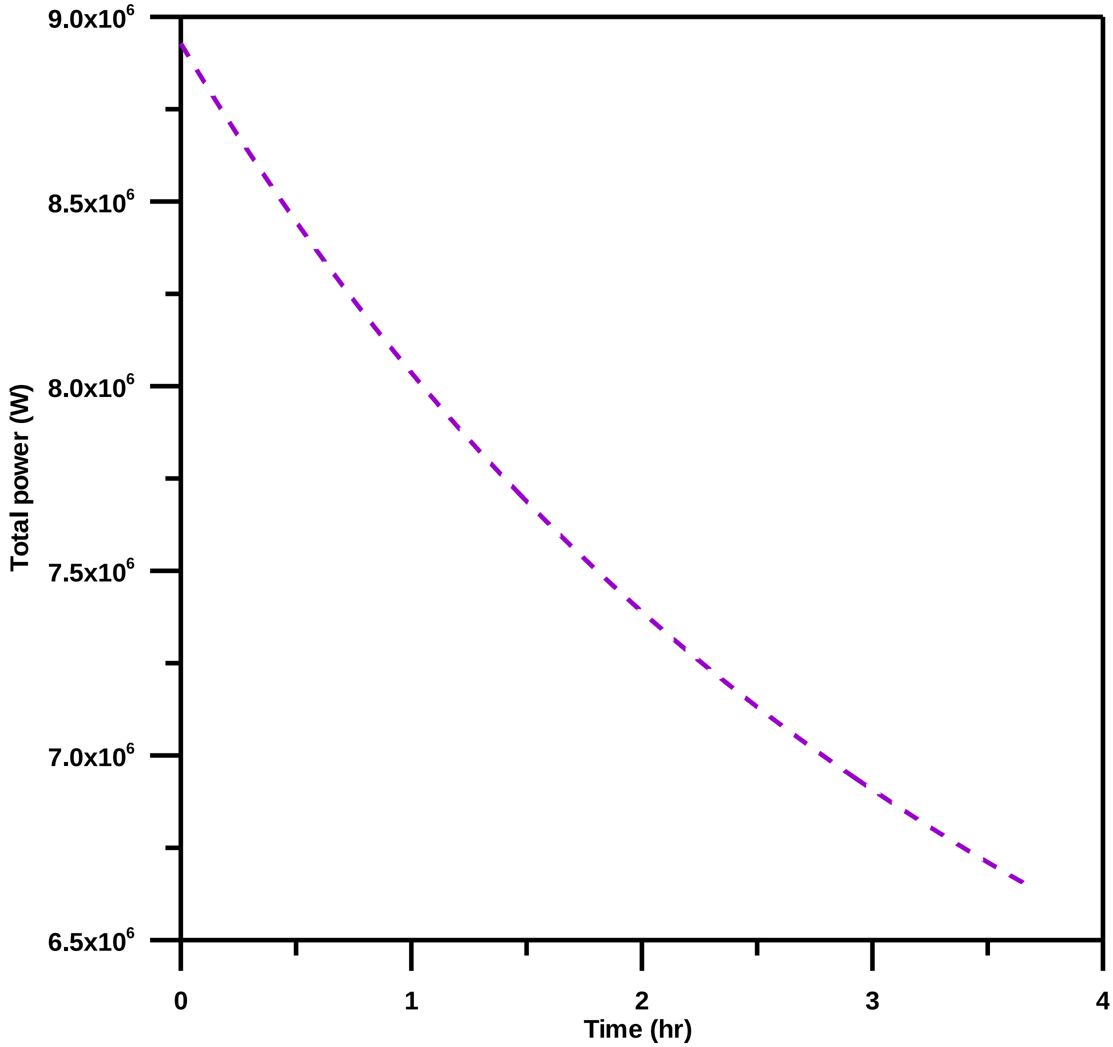


Figure 9 Total Power

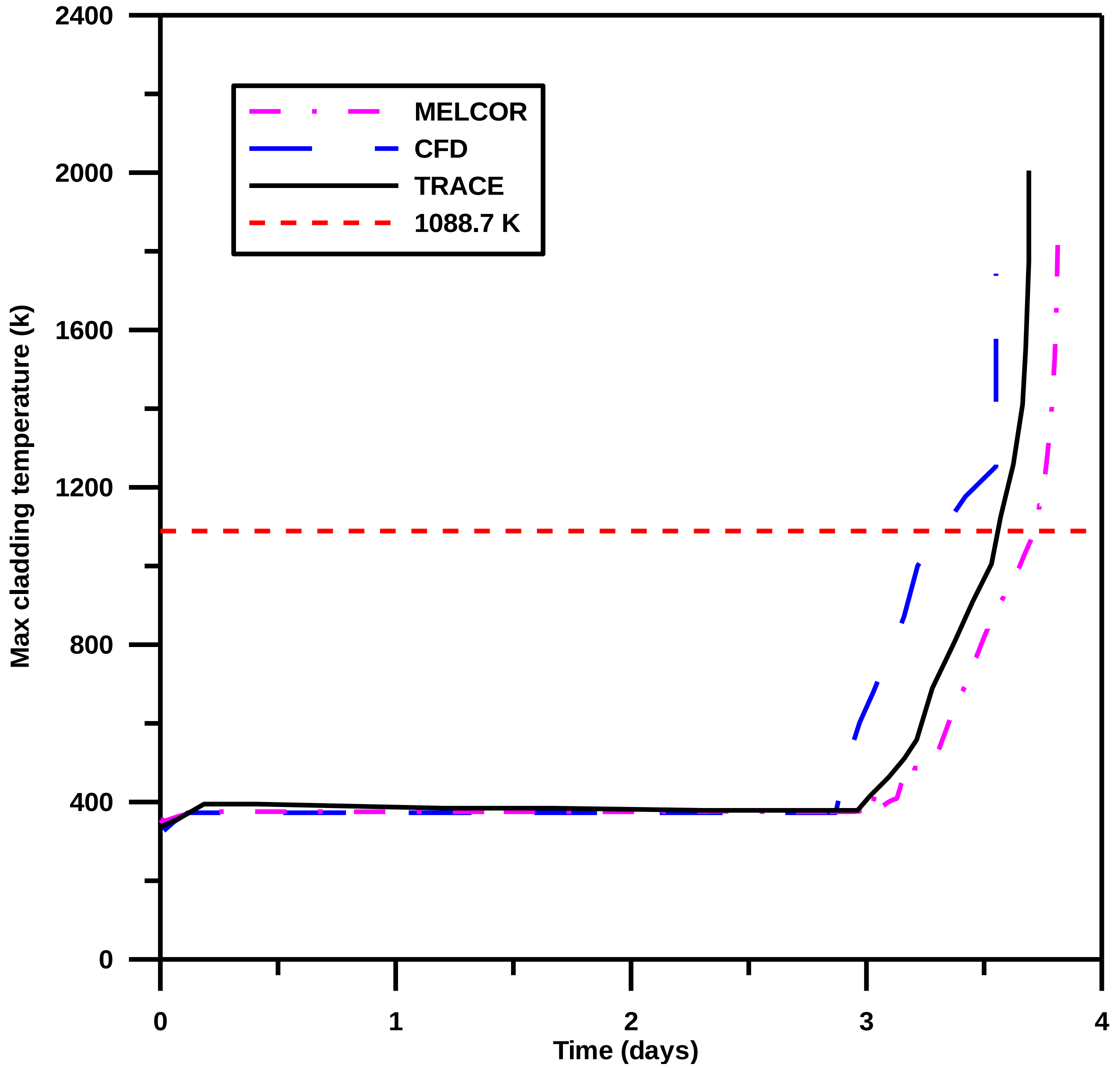


Figure 10 The Max Cladding Temperature Results

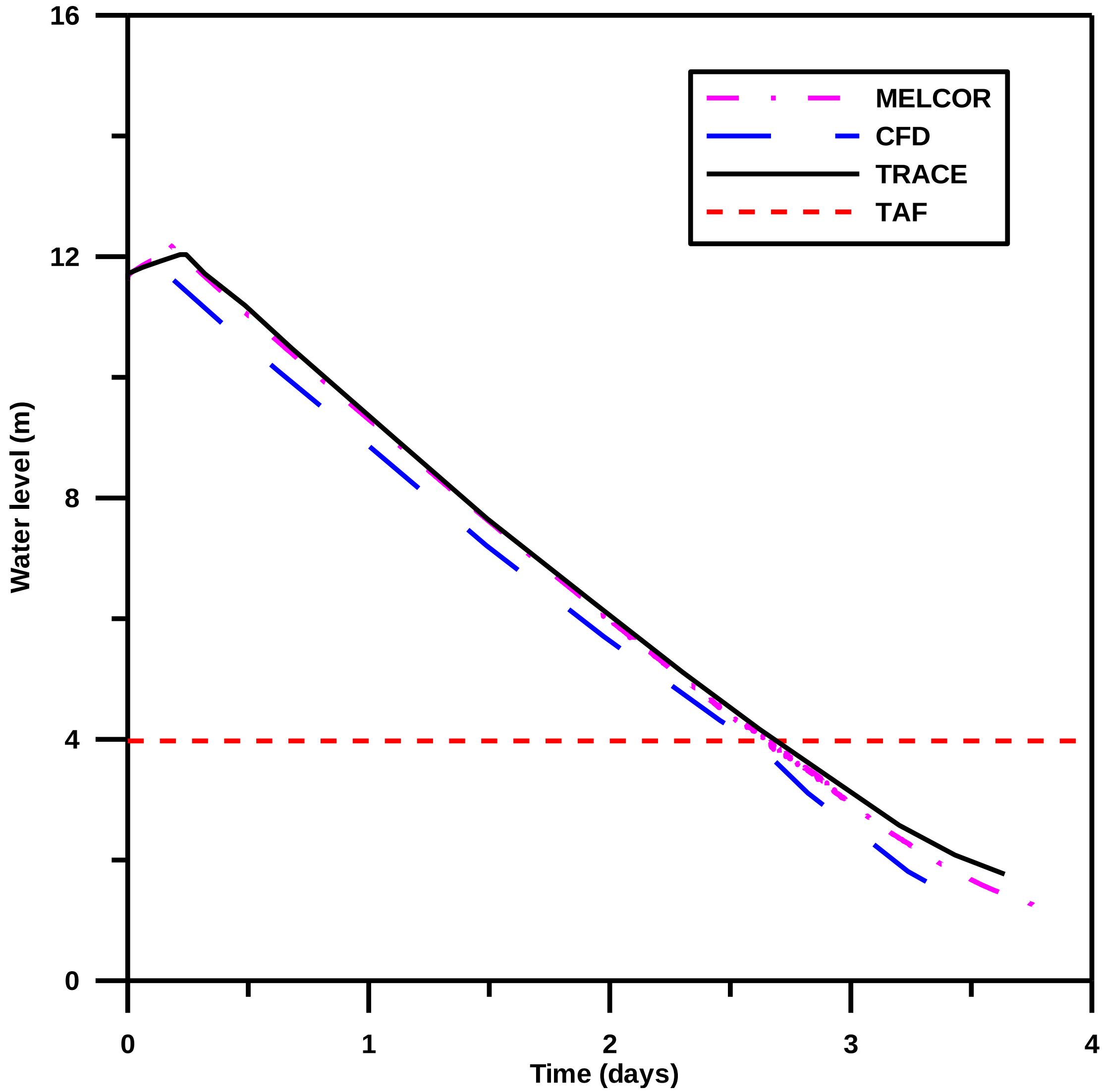


Figure 11 **The Water Level Results**

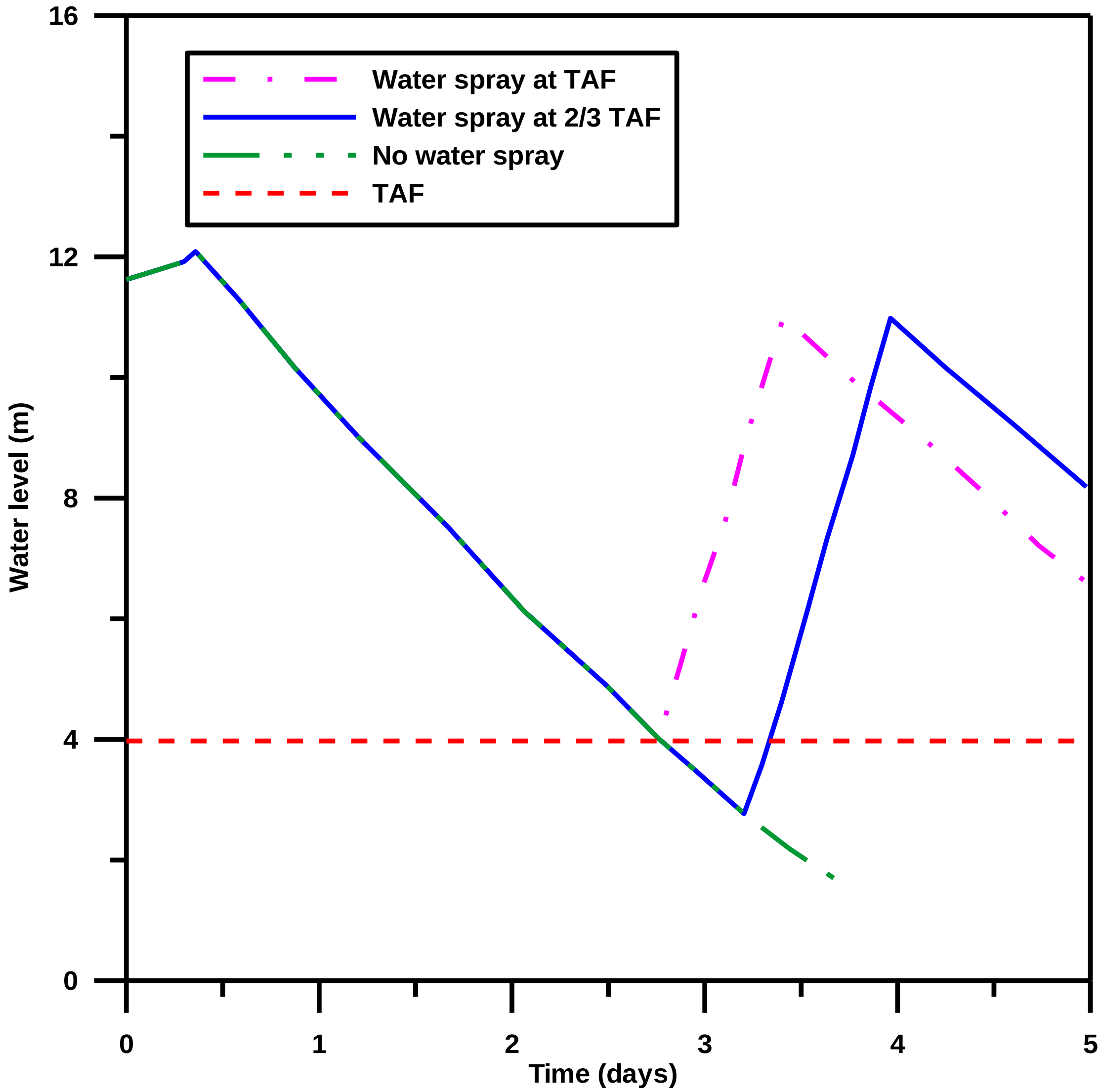


Figure 12 The Water Level Results for the Water Spray Cases

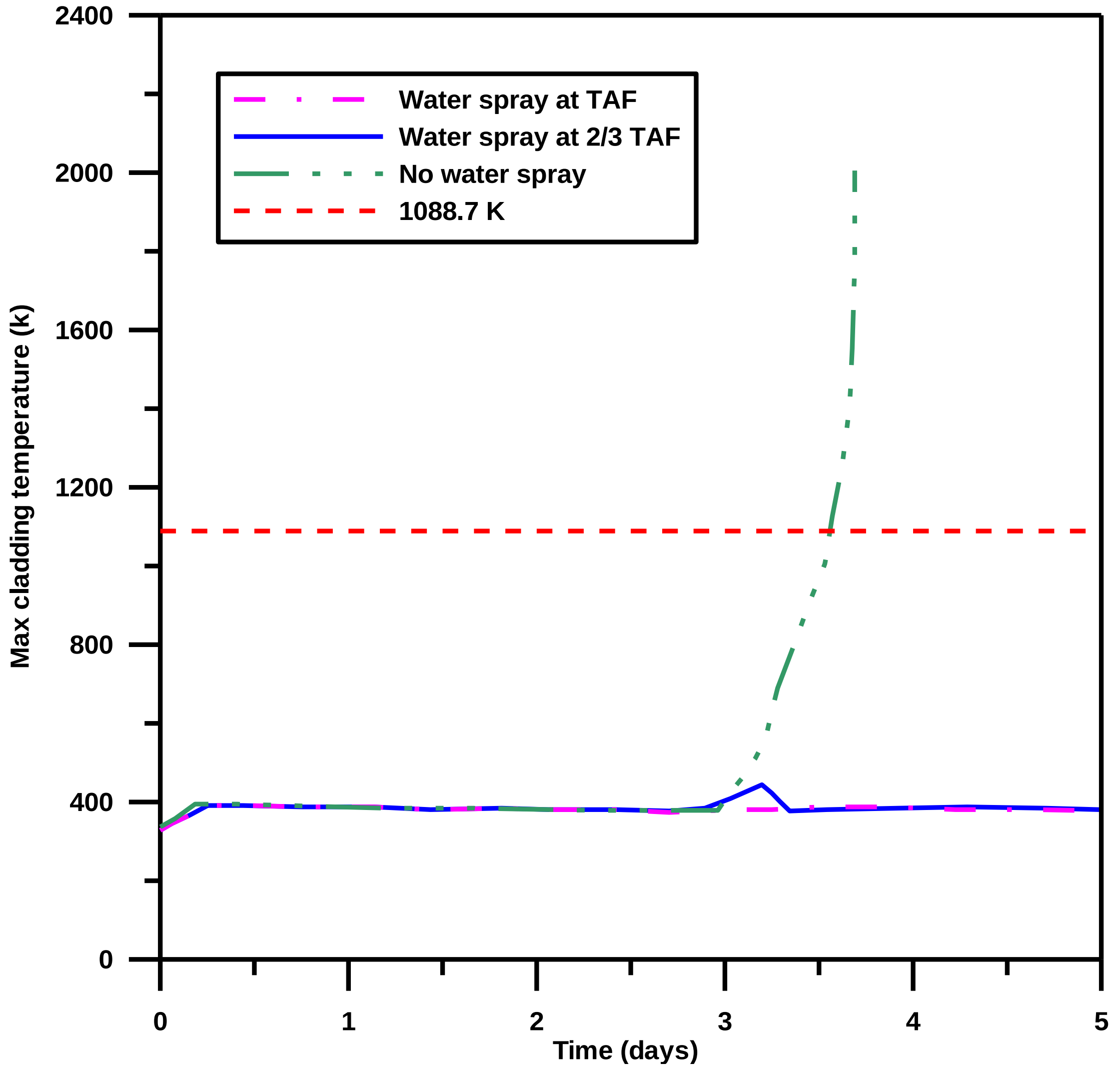


Figure 13 The Max Cladding Temperature Results for the Water Spray Cases

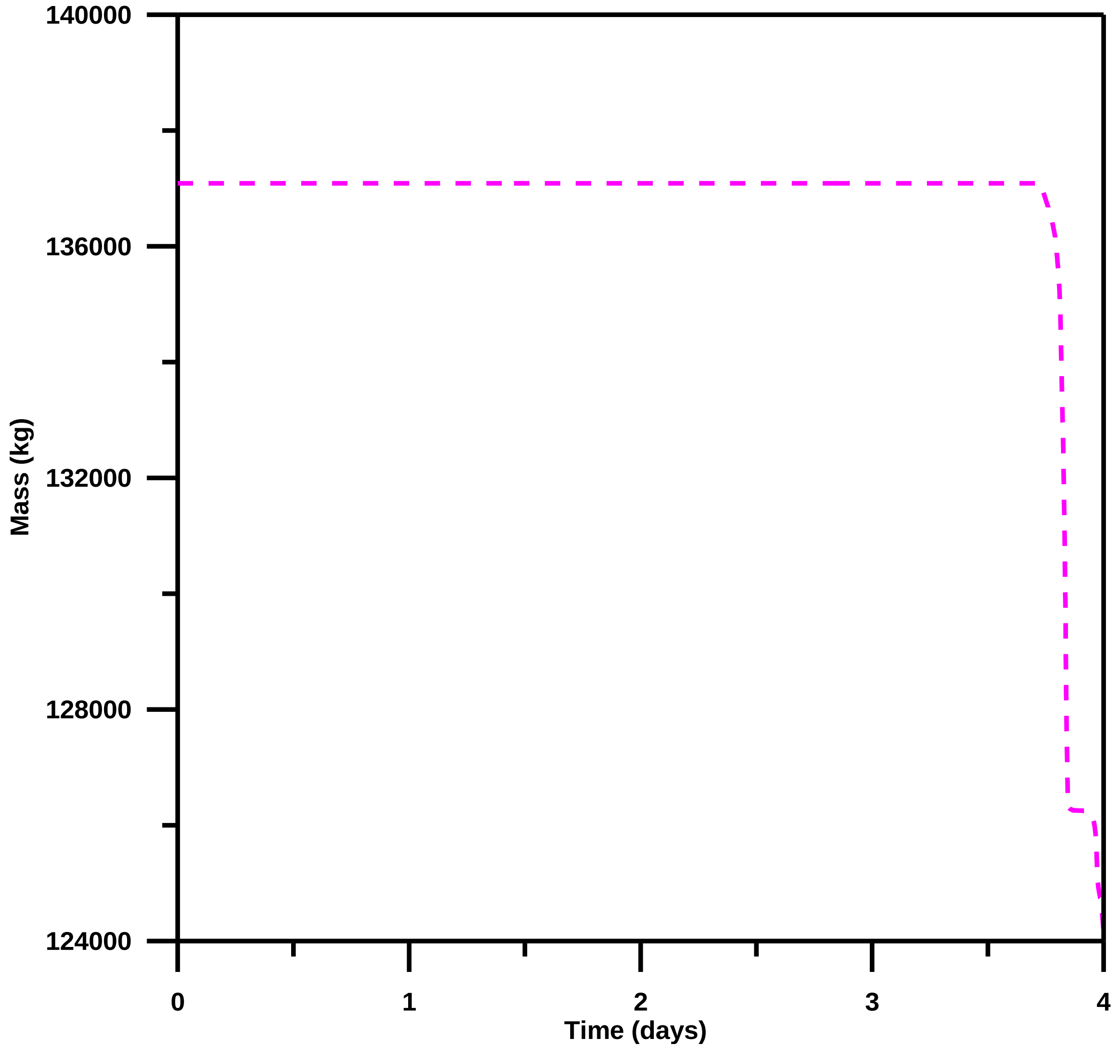


Figure 14 Total Zr Mass Result

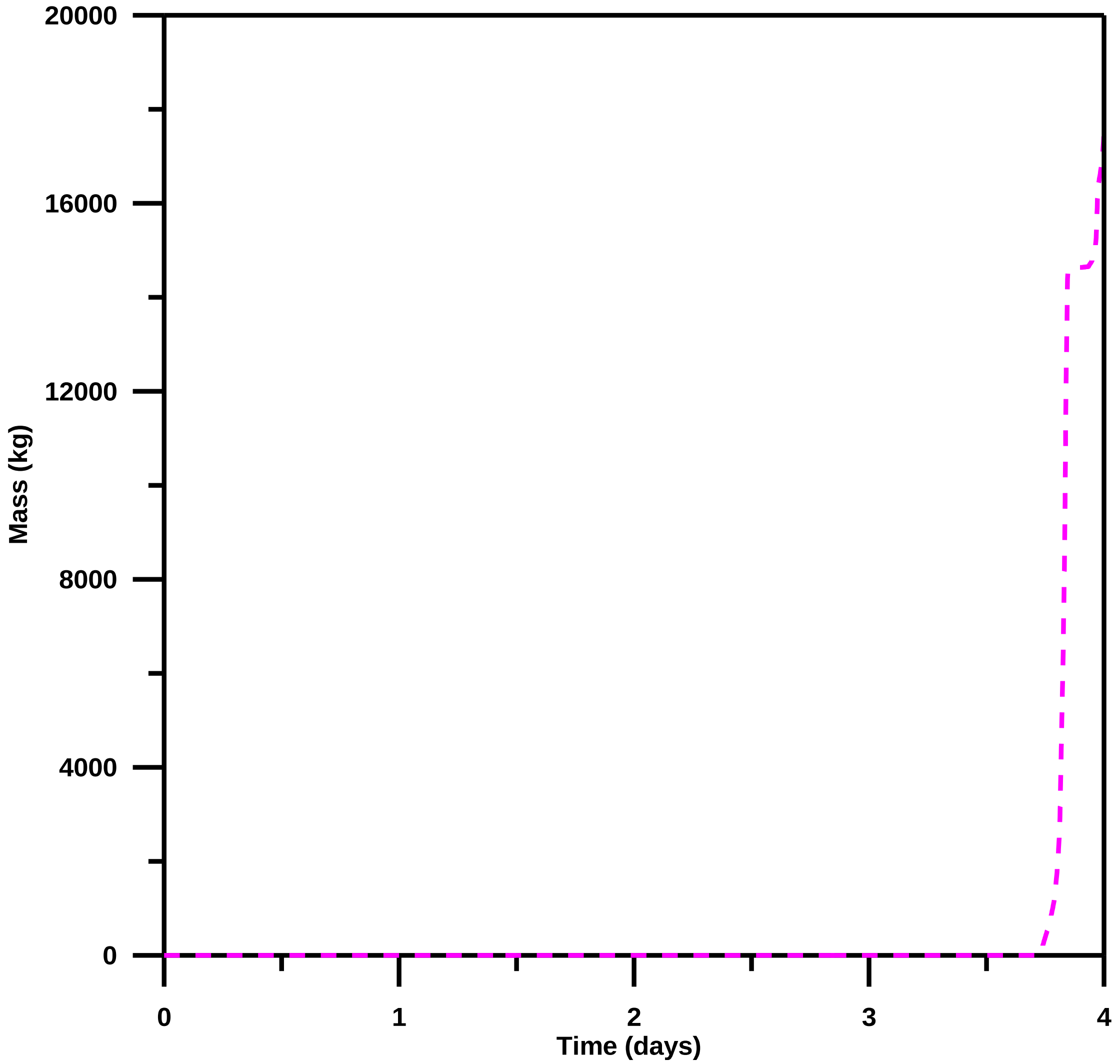


Figure 15 **Total ZrO_2 Mass Result**

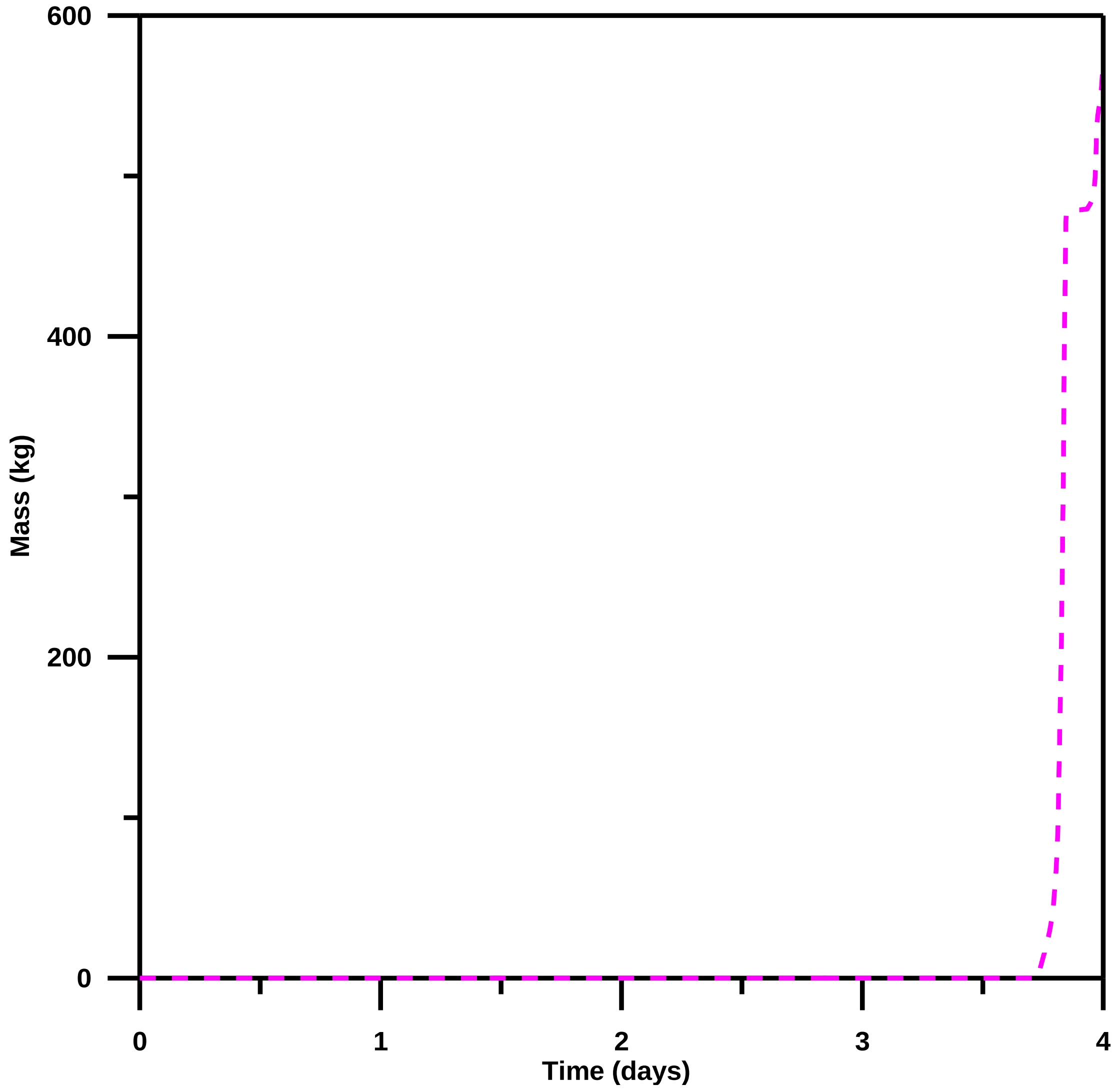


Figure 16 **Total H_2 Mass Result**

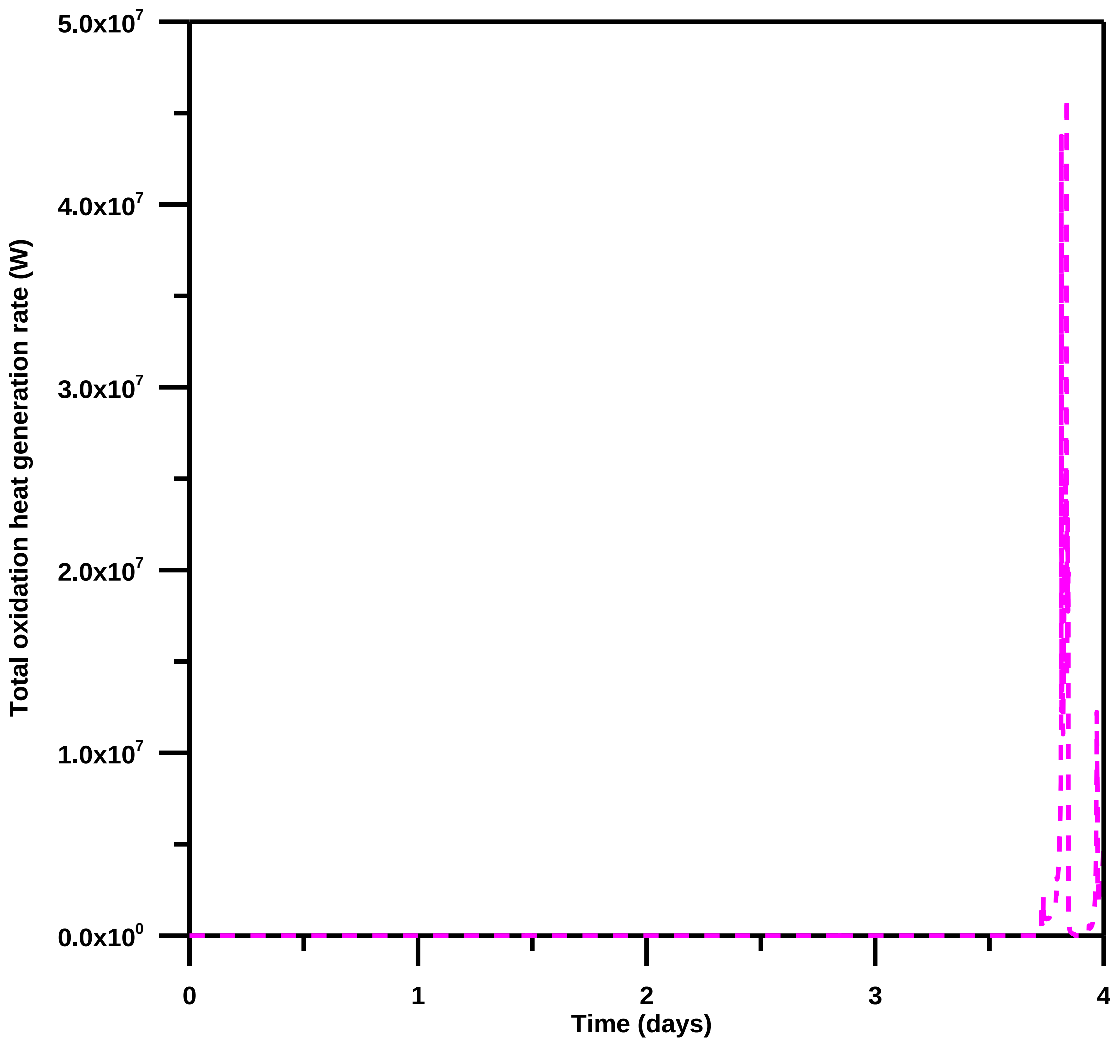


Figure 17 Total Oxidation Heat Generation Rate Result

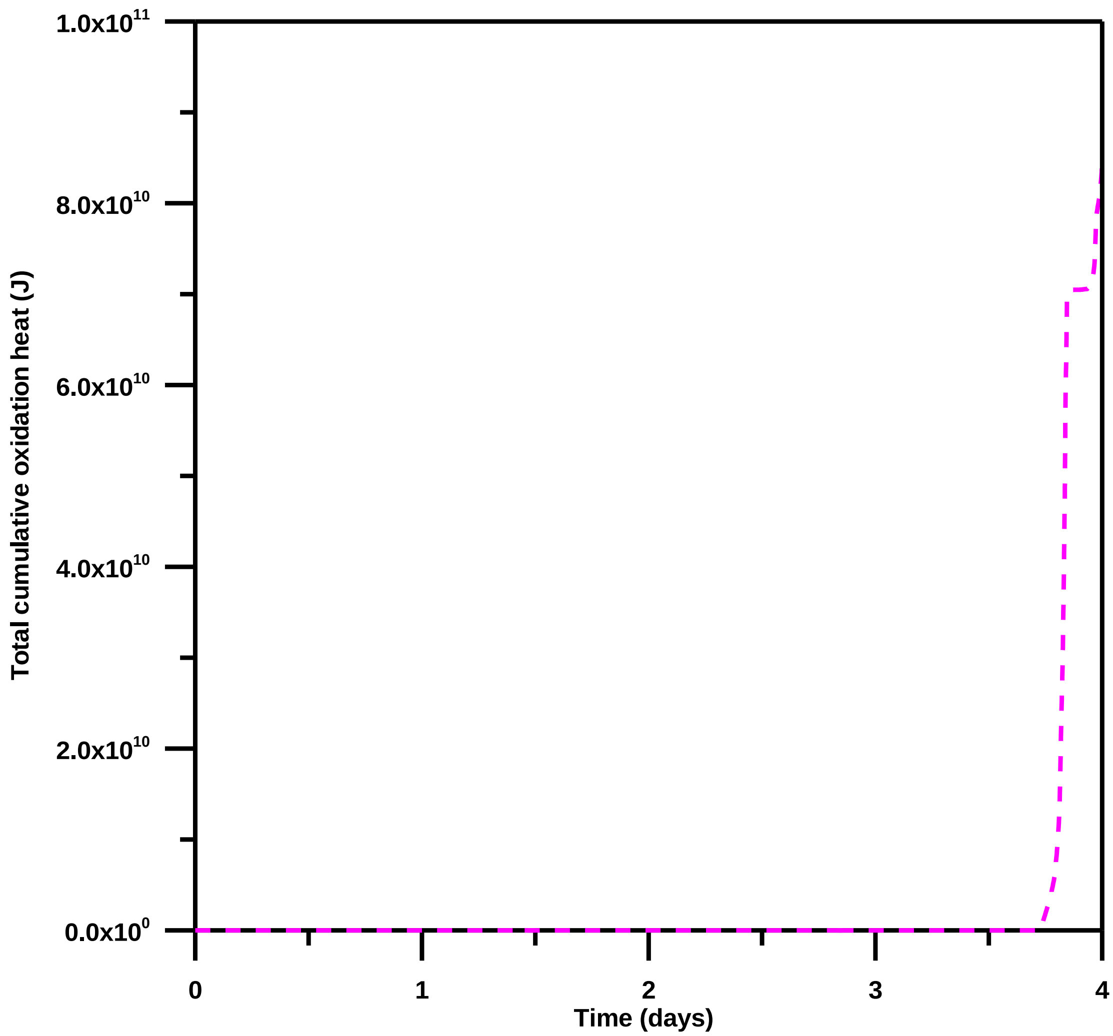


Figure 18 Total Cumulative Oxidation Heat Result

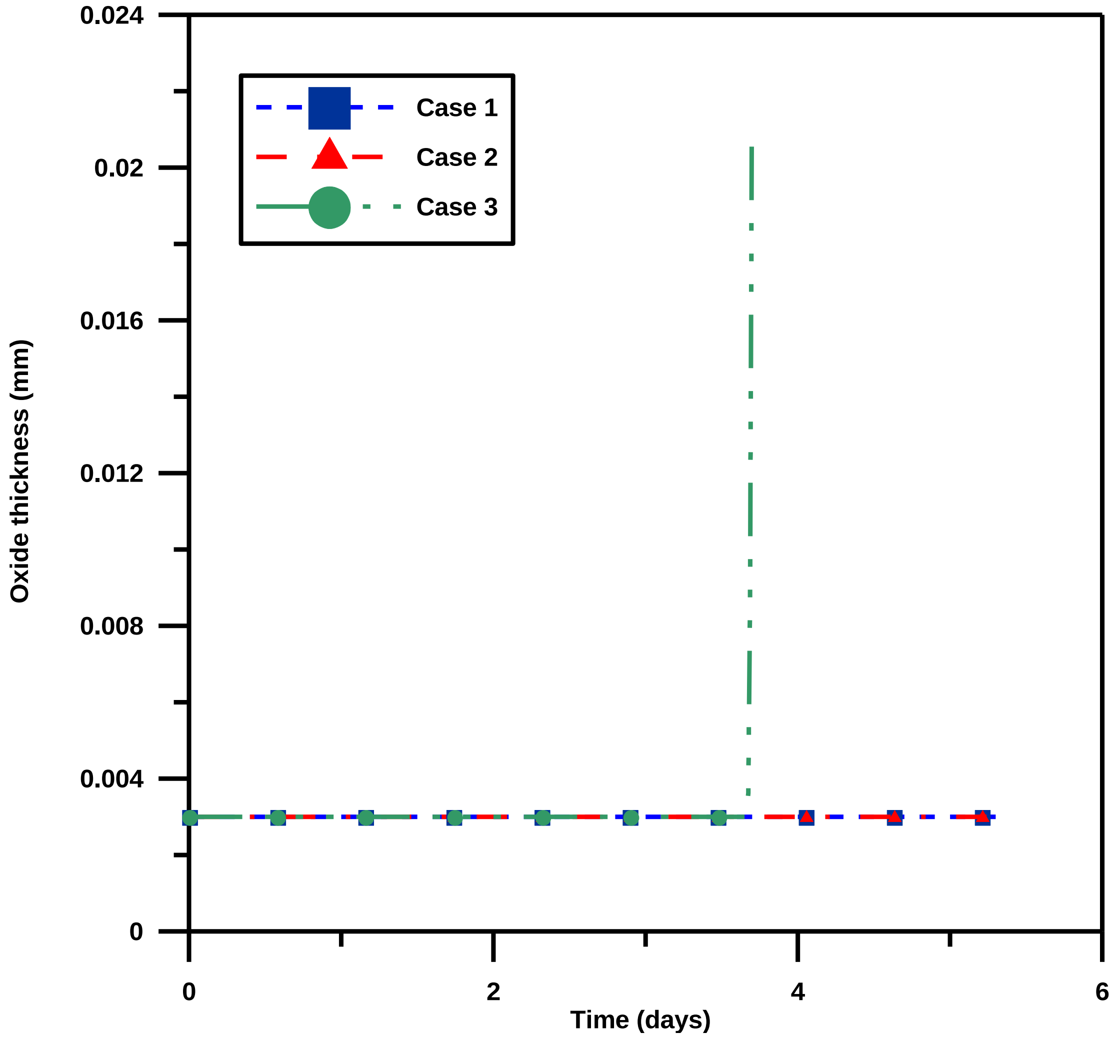


Figure 19 The Oxide Thickness Results

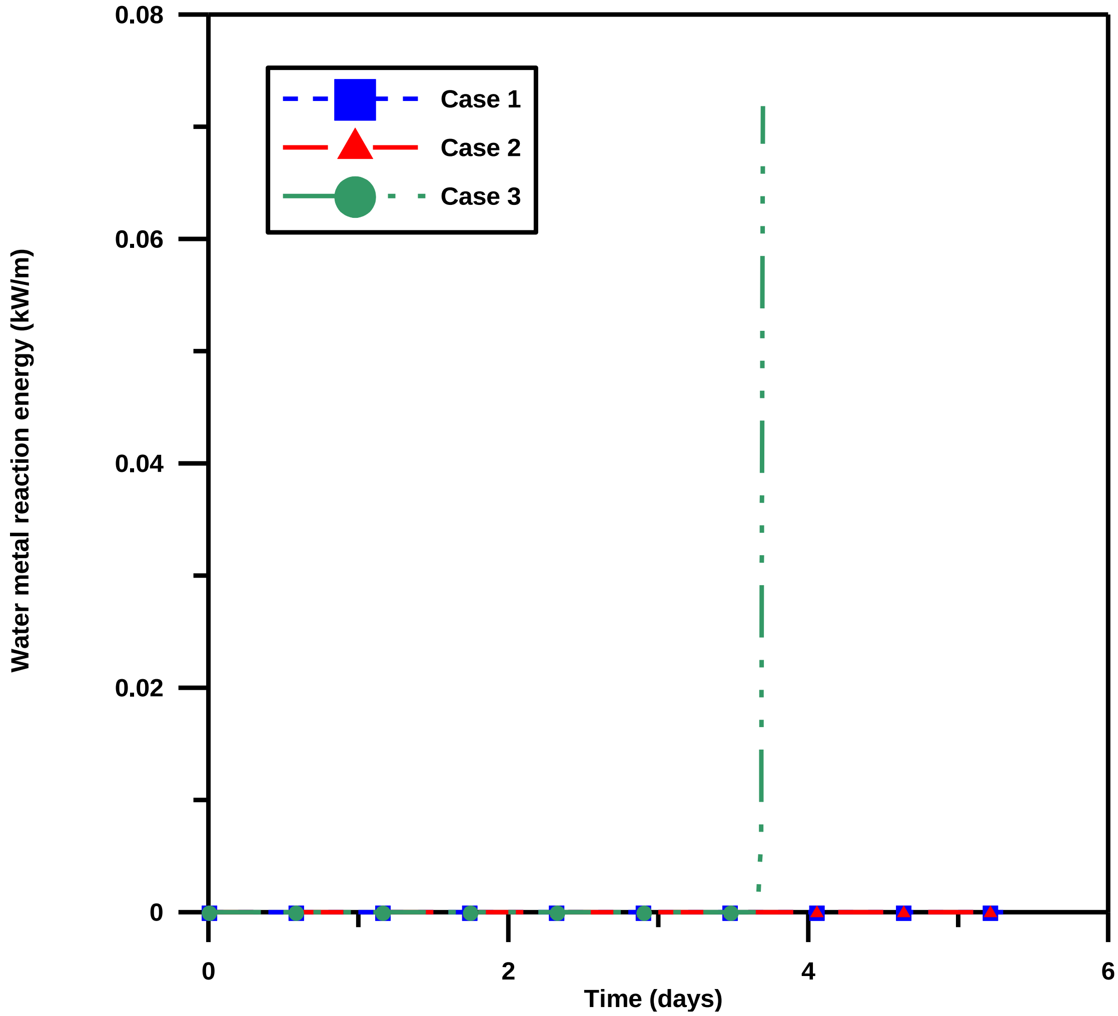


Figure 20 The Water Metal Reaction Energy Results

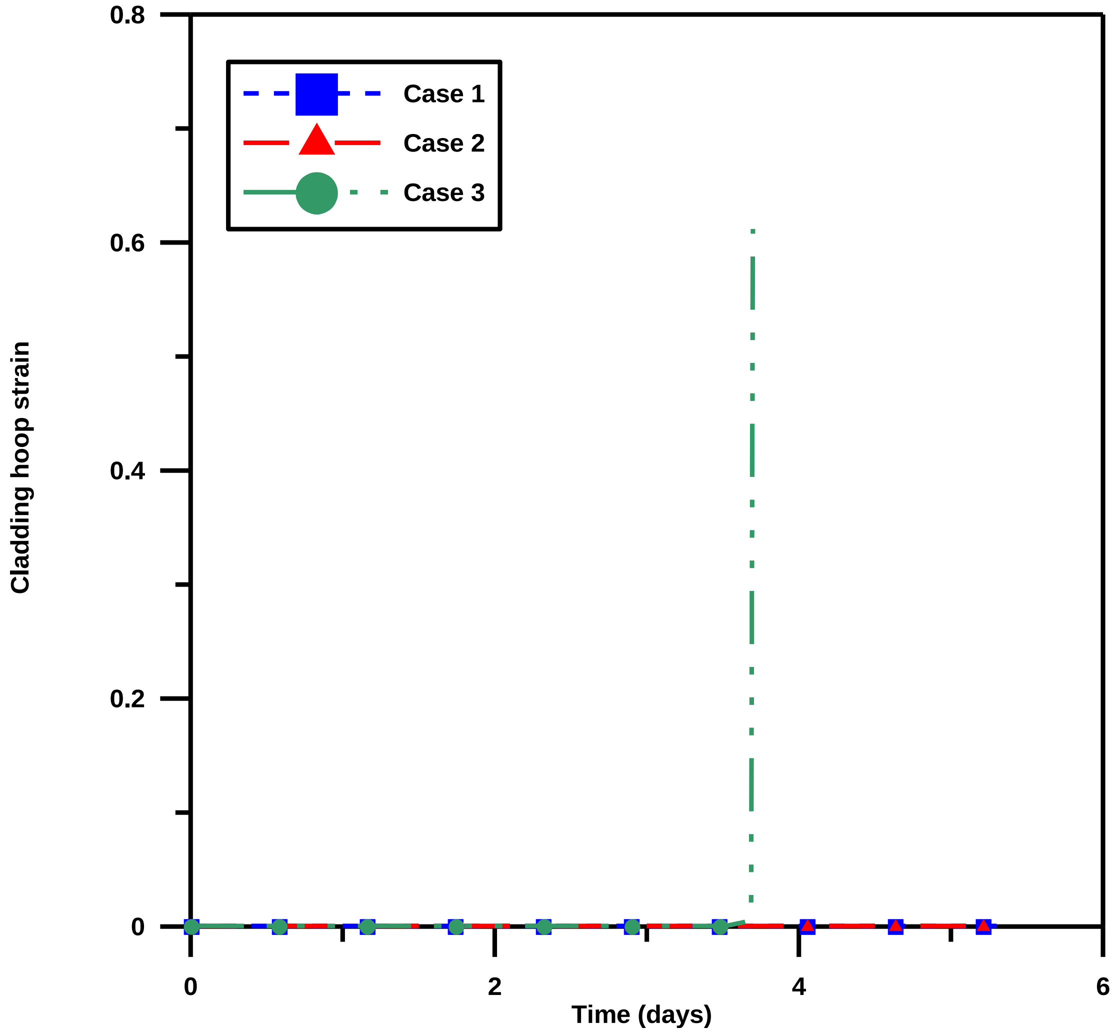


Figure 21 The Cladding Hoop Strain Results

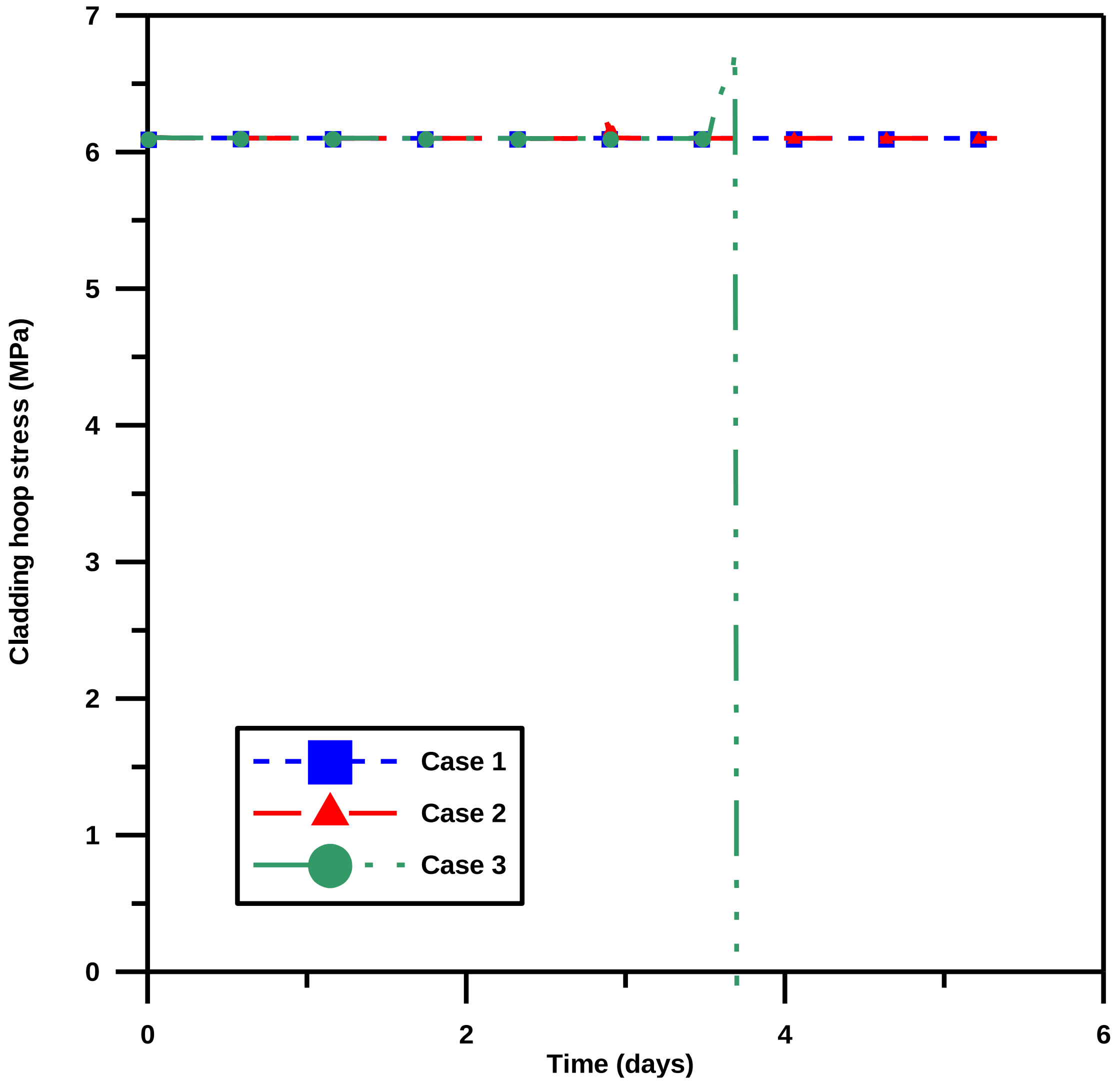


Figure 22 The Cladding Hoop Stress Results

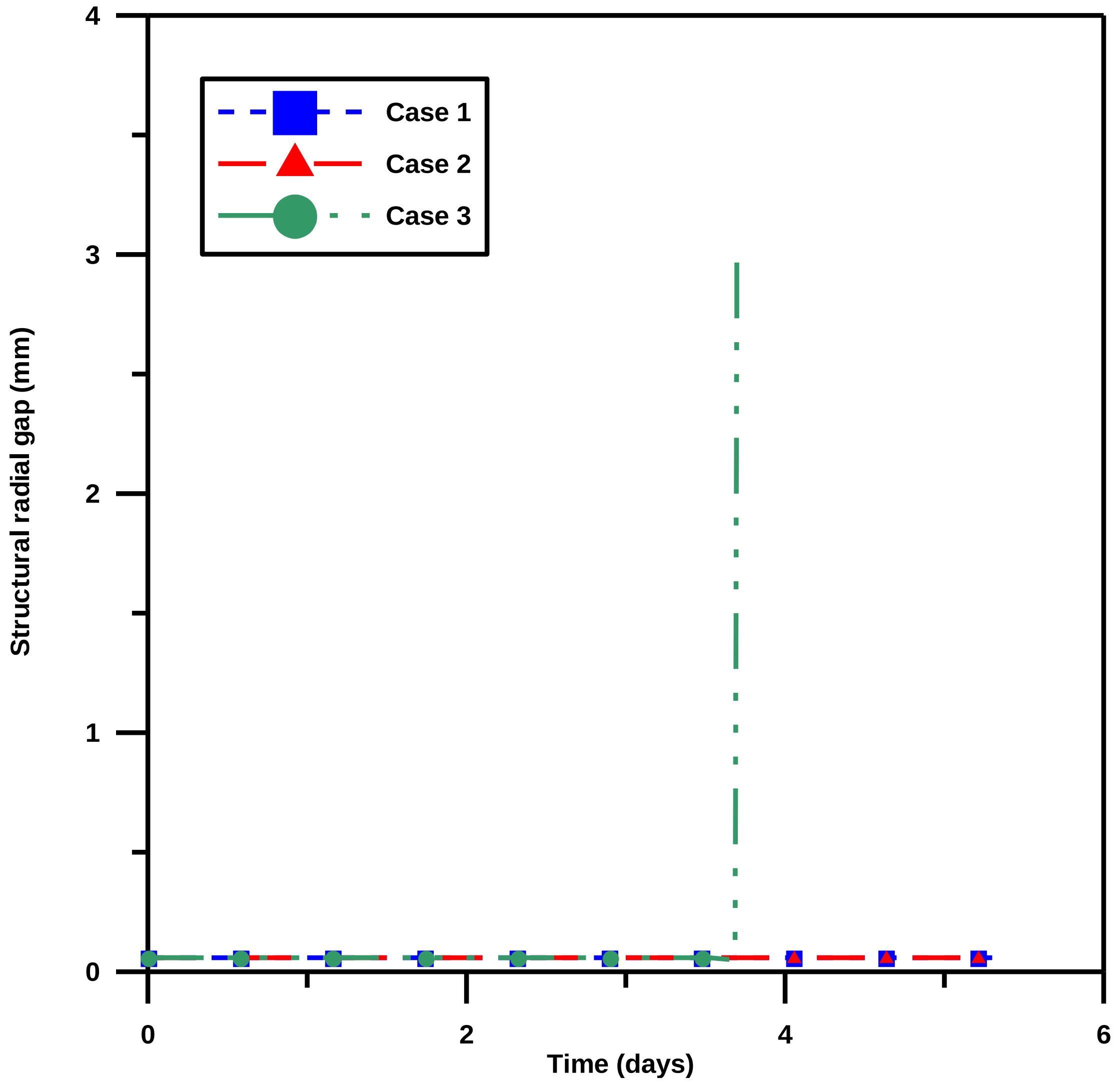
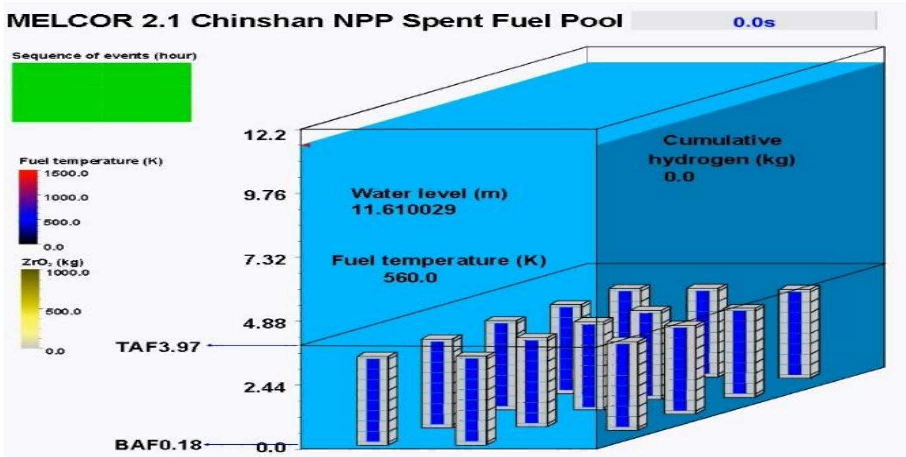
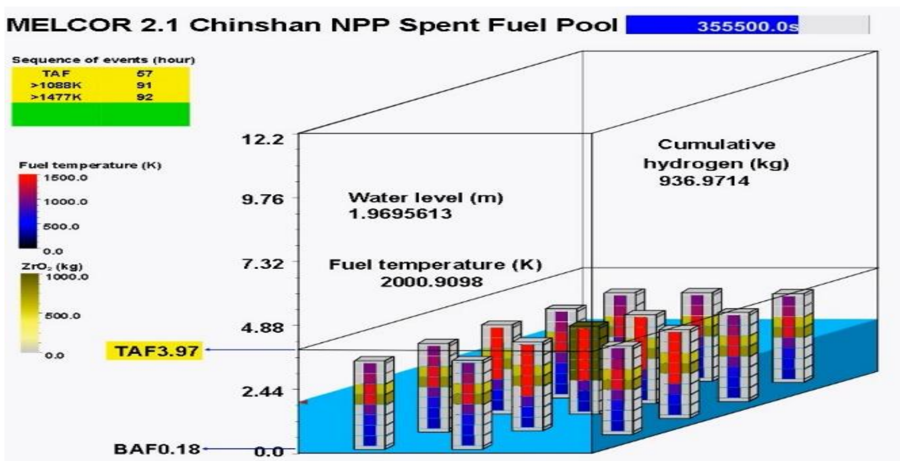


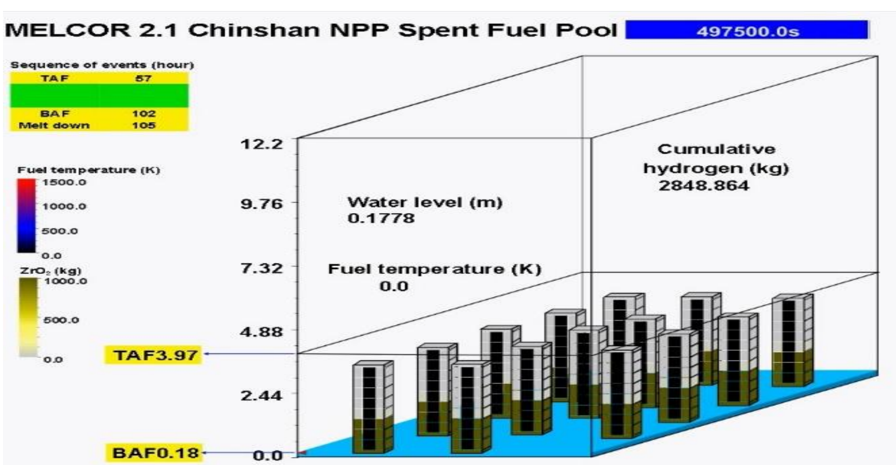
Figure 23 The Structural Radial Gap Results



(a) 0 sec



(b) 355500 sec



(c) 497500 sec

Figure 24 The Animation Model (a) 0 sec, (b) 355500 sec, (c) 497500 sec

4 CONCLUSIONS

This study was established TRACE, MELCOR, CFD, and FRAPTRAN models of Chinshan NPP SFP successfully. The SFP safety analysis was performed under the cooling system failure condition using the above models. The analysis results (max cladding temperature and water level) of TRACE, MELCOR, and CFD were similar in this case. TRACE results depict that the uncovered of the fuels presented at 2.7 days and the cladding temperature reached 1088.7 K at 3.6 days after the cooling system failed. The sensitive study of the time point for water spray was also performed. The results illustrate that the cladding temperature can be kept under 1088.7 K if the water spray is performed before the water level is lower than 2/3 TAF. Additionally, using FRAPTRAN and TRACE's results performed the fuel rod performance analysis in this study. According to FRAPTRAN results, the highest cladding temperature located on the node 21 of the fuel rod. It also indicates that the cladding burst roughly at 3.7 days for the case under the cooling system failure condition. In addition, the results of FRAPTRAN depicted that the fuel rod performance was affected by the variation of the water level and max cladding temperature.

5 REFERENCES

U.S. NRC, TRACE V5.840 user's manual, 2014.

U.S. NRC, TRACE V5.0 assessment manual, 2010.

Sandia National Laboratories, MELCOR Computer Code Manual, 2011.

K.J. Geelhood, W.G. Luscher, C.E. Beyer, and J.M. Cuta, FRAPTRAN 1.4: A Computer Code for the Transient Analysis of Oxide Fuel Rods, NUREG/CR-7023, Vol. 1, 2011.

R. Gauntt, D. Kalinich, J. Cardoni, J. Phillips, A. Goldmann, S. Pickering, M. Francis, K. Robb, L. Ott, D. Wang, C. Smith, S. St.Germain, D. Schwieder, C. Phelan, Fukushima Daiichi Accident Study, SANDIA REPORT (Status as of April 2012), SAND2012-6173, 2012.

J.J. Carbajo, MELCOR Model of the Spent Fuel Pool of Fukushima Dai-ichi Unit 4, American Nuclear Society Meeting, Chicago, Illinois, USA, 2012.

N.E. Könönen, Spent Fuel Pool Accidents in A Nordic BWR, Proceedings of the 2013 21st International Conference on Nuclear Engineering (ICONE21), Chengdu, China, 2013.

Y.S. Tseng, et al., Analysis of Alternative Shutdown Cooling for Chinshan Nuclear Power Plant Using CFD Simulation, INER report, INER-7700, 2010.

NEI, B.5.b Phase 2 & 3 Submittal Guideline, NEI 06-12 Revision 2, 2006.

U.S. NRC, Standard review plan, NUREG-0800, 2007.

Taiwan Power Company, Chinshan nuclear power plant ultimate response guideline, No. 1451, 2012.

NRC FORM 335 (12-2010) NRCMD 3.7 — U.S. NUCLEAR REGULATORY COMMISSION	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)
BIBLIOGRAPHIC DATA SHEET *(See instructions on the reverse)*	**NUREG/IA-0482**

2. TITLE AND SUBTITLE

Using TRACE, MELCOR, CFD, and FRAPTRAN to Establish the Analysis Methodology for Chinshan Nuclear Power Plant Spent Fuel Pool

3. DATE REPORT PUBLISHED

MONTH	YEAR
October	**2018**

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

Jong-Rong Wang, Chunkuan Shih, Shao-Wen Chen, Yung-Shin Tseng, Yu Chiang, Wan-Yun Li, Jung-Hua Yang, Hsiung-Chih Chen, Show-Chyuan Chiang*, Tzu-Yao Yu*

6. TYPE OF REPORT

Technical

7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Nuclear and new Energy Education and Research Foundation
Institute of Nuclear Engineering and Science,
National Tsing Hua University
101 Section 2, Kuang Fu Rd., HsinChu, Taiwan

*Department of Nuclear Safety, Taiwan Power Co.
242, Section 3, Roosevelt Rd., Zhongzheng Dist.

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

Division of Systems Analysis
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

K. Tien, NRC Project Manager

11. ABSTRACT (200 words or less)

Chinshan nuclear power plant (NPP) is the first plant in Taiwan and is a BWR/4 plant. People have more concern for the NPPs safety in Taiwan after Fukushima NPP disaster happened. Hence, we established the analysis methodology and performed the safety analysis of Chinshan NPP SFP (spent fuel pool) by using TRACE, MELCOR, CFD, and FRAPTRAN codes. There were two steps in this study. The first step was the establishment of Chinshan NPP SFP models by using TRACE, MELCOR, and CFD. Then, under the SFP cooling system failure condition (Fukushima-like accident), the transient analysis was performed. Additionally, the sensitive study of the time point for water spray was also performed. The next step was the analysis of the fuel rod performance by using FRAPTRAN and TRACE analysis results. Finally, the animation model of Chinshan NPP SFP was presented by using the SNAP animation function with MELCOR's results.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)

Spent Fuel Pool (SFP)
Fukushima-like Events
TRACE/SNAP/FRAPTRAN
Methods for Estimation of Leakages and Consequences of Release (MELCOR)
Computational Fluid Dynamics (CFD)
Chinshan Nuclear Power Plant (NPP)

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page) unclassified

(This Report) unclassified

15. NUMBER OF PAGES

16. PRICE

NUREG/IA-0482

Using TRACE, MELCOR, CFD, and FRAPTRAN to Establish the Analysis Methodology for Chinshan Nuclear Power Plant Spent Fuel Pool

October 2018