第14回原子力委員会
参考資料第2-2号

過酷事故に関する実験研究及び解析コードリスト(作業中　2017/12/15)

分類
①　炉心損傷初期(再冠水による過熱炉心の損傷も含む)
②　炉心崩壊期【炉心損傷後期】ヨウ素やセシウムを含むFP(エアロゾル、ガス状など)や水素、過熱蒸気などの挙動を含む
③　原子炉容器内溶融物挙動と冷却性(IVRを含む)
④　原子炉容器と1次冷却系統破損・溶融物・FPの格納容器内への放出挙動
⑤　格納容器直接加熱(DCH)など格納容器と貫通部の破損挙動(PWRの蒸気発生器伝熱管の破損による格納容器バイパスを含む)
⑥　溶融物の格納容器床での広がり挙動、構造物や格納容器内壁との相互作用
⑦　溶融物とコンクリートの相互作用(MCCI)
⑧　原子炉1次系からの揮発性FP等の放出に影響する挙動、配管沈着挙動等も含む
⑨　格納容器と原子炉建屋からの揮発性FP等の放出に影響する挙動
⑩　原子炉建屋外での揮発性・気体状FP等の拡散と付着挙動
⑪　オフサイト環境中でのFP等の挙動
⑫　過酷事故時のRCICなど非常用炉心冷却系や原子炉1次系と格納容器とその弁、シール等の挙動
⑬　個別事象
（ア）溶融物・水相互作用【水蒸気爆発】【原子炉容器内、原子炉容器外】
（イ）水素燃焼・爆発
（ウ）再臨界
⑭　設備設計関係　⑫と統合てもよいかもしれない。
（ア）原子炉容器内溶融物の原子炉容器外冷却(AP1000)
（イ）コアキャッチャ(EPR,　VVER)
（ウ）フィルターベント、サプレッションプールFP挙動
（エ）水素再結合器
（オ）格納容器スプレイ等
（カ）使用済燃料プール
⑮　解析コード

出典
1 原子力委員会技術等検討小委員会第3回資料第2号 「事故時ソースターム」平成23年10月25日 松村健
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2 SARnet Scope and Objectives
http://www.sar-net.eu/?q=node/1
3 Kalsruhe Institute of Technology web. Nuclear Safety Research Program
http://nuklear-server.nuklear.kit.edu/lacomeco/Facilities/facilities.htm
4 Overview of Light Water Reactor Severe Accident Analysis and Test Programs at Argonne Mitchell T.Farmer
International Workshop On New Horizons in Nuclear Ractor Thermal Hydraulics and Safety Mumbai,India Jan. 13-15,2014
5 海外における溶融燃料挙動に関する研究の現状 原子力機構 永瀬文久2013年原子力学会 春の大会
6 Overview of Severe Accident Reseach Activities at KIT A.Miassoedov
7 PWR規制庁新規制基準適合性審査資料 第58回（H25.12.17） 資料2-1他
https://www.nsr.go.jp/disclosure/committee/yuushikisya/tekigousei/power_plants/h25fy/20131217.html
8 BWR規制庁新規制基準適合性審査資料
9 エネ総工研内藤氏資料
10 TRENDS IN SEVERE ACCIDENT RESEARCH IN EUROPE: SARNET NETWORK FROM EURATOM TO NUGENIA
https://www.euronuclear.org/events/topsafe/topsafe2017/pdf/fullpapers/TopSafe2017-A0027-fullpaper.pdf
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15 http://s538600174.onlinehome.fr/nugenia/portfolio/air-sfp/
16 NEA Melt Coolability and Concrete Interaction (MCCI) Project
http://www.oecd-nea.org/jointproj/mcci.html
17 NEA Sandia Fuel Project (SFP)
http://www.oecd-nea.org/jointproj/sfp.html
18 IN-VESSEL CORE DEGRADATION CODE VALIDATION MATRIX Update 1996-1999 NEA/CSNI/R(2000)21
19 Implementation of Severe Accident Management Measures, ISAMM 2009Workshop Proceedings, Vol. I NEA/CSNI/R(2010)10/PART1

研究項目分類	プロジェクト名	試験内容	実施機関	年開始	重要度	出典	報告書等
①	LOFT	INLのPWR(LOFT炉) を用いて事故時の熱水力挙動、ECCSの有効性確認、FPの移行挙動を確認 The Loss of Fluid Test Facility (LOFT) was first conceived by the USAEC in 1962.Following the completion of the facility in 1976, a series of thermal hydraulic-type tests was carried out under the sponsorship of the USNRC, to study large and small break accidents.	米国INL	1962		18	
①	MAESTRO	燃料-被覆管等の高温物性	仏IRSN			5	
①	MOZART	空気中でのZr酸化	仏IRSN			5	
①	ＣＡＢＲＩ	CABRI炉内試験プロジェクトの当初の主要課題は高速炉の「仮想的炉心崩壊事故 (Hypothetical Core Disruptive Accident) 」の挙動解明	仏				
①	管群ボイド		NUPEC			7	
①	THTF	Thermal Hydraulic Test Facilityで実施された、PWRの燃料を模擬した、LOCA時の伝熱流動特性実験	米国ORNL			7	
①	NIELS	Initial investigation of early phase melt progression, including effect of PWR absorber materials, in a single rod and small bundle environment. There were 12 basic single rod tests (ESSI series); 3 additional single rod tests to study the effect of hydrogen (ESA series), 2 3x3 bundle tests with no absorber (ESBU series) and 6 3x3 bundle tests including PWR absorber (ABS series).	独 KfK Karlsruhe	1982		9,18	
①	CORA	UO2燃料、制御棒チャンネルボックスを電気加熱 To investigate out-of-pile the early phases of core degradation in light water reactor systems (PWR and BWR) in a bundle environment. The series was extended (2 further tests) to examine VVER phenomena.	(FZ Karlsruhe, formerly KfK, Germany) 1987 - 1992 (VVER tests to 1993)	1987		1､9	
①	PHEBUS SFD	Investigate in-pile early phase of PWR core degradation in a bundle environment. Experimental data base and odelling of SFD processes in ICARE2 code.	仏(IPSN/CEA - Cadarache 1986 - 1989	1986		9,18	

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①	PBF-SFD	The specific objectives of the PBF-SFD series of tests were to: (1) investigate fuel rod damage following severe cladding oxidation, melt relocation, and fuel rod fragmentation; (2) measure the release rates, transport, deposition of fission products; (3) determine the magnitude and timing of hydrogen generation; (4) investigate the coolability of test bundles following various types of damage; (5) determine the behavior of irradiated fuel rods compared with fresh fuel rods and to evaluate the effects of control rods.	(Power Burst Facility - Severe Fuel Damage), Idaho National Engineering Laboratory (INEL), Idaho Falls, USA 1982 - 1985	1982		9,18	
①	CBDP	Full-Length High Temperature (FLHT) Tests of the Coolant Boilaway Damage Progression (CBDP) Program were conducted by Pacific Northwest Laboratory (PNL) in the NRU reactor at AECL Chalk River, Ontario, Canada	カナダ　AECL			9	
①	ACRR-ST	照射済燃料、FPの影響 Fission product release tests conducted in the Annular Core Research Reactor Determine effects of temperature, pressure, fuel damage state, gas environment (reducing or oxidizing), clad oxidation state and fuel clad geometry on magnitude and rate of release of fission products from previously irradiated fuels. The scope of the test programme was reduced to the two tests performed.	米国SNL 1986 - 1989	1986		5,9,18	
①	ACRR-DF	原子炉ACRRを用いてUO2,制御棒等を核加熱 Fuel damage tests conducted in the Annular Core Research Reactor To investigate in-pile using prototypic materials the early phases of core degradation in light water reactor systems (PWR and BWR) in a bundle environment.	米国SNL			1,9,18	
①	LOFT LP-FP	LOFT is a major research programme generally aimed at studying the behaviour of a PWR primary system under LOCA situations. OECD sponsored LOFT after 1983 and decided to undertake an additional matrix of eight tests, of which the last two tests should include fuel damage and fission product release.	米国INEL 1976-1982 Period Sponsored by USNRC, 1983-1986 Period Sponsored by OECD	1976		9,18	

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①	CODEX	To investigate early-phase core degradation in light water reactors (Western LWR and VVER designs). The series of 5 tests to date has investigated the effects of quench in VVER geometry, and the effect of of air ingress on pre-oxidised Western LWR bundles.	ハンガリーAEKI,	1995		9,18	
①	NRU-FLHT	Between 1985 and 1987, four full-length high temperature (FLHT) tests of the Coolant Boilaway and Damage Progression (CBDP) Program were conducted by PNL in the NRU reactor at Atomic Energy of Canada Ltd. (AECL) Chalk River using highly instrumented and insulated assemblies of 12 full-length (3.7 m) light water reactor (LWR) fuel rods. The objectives of the CBDP Programme were to: · obtain data for evaluating the effects of coolant boilaway and core damage progression in an LWR; and · investigate integral severe accident phenomena along a full-length bundle.	カナダAECL	1985		18	
①	OECD/NEA Sandia Fuel	This project provided experimental data relevant for hydraulic and ignition phenomena of prototypic LWR fuel assemblies. The proposed experiments focused on thermal-hydraulic and ignition phenomena in pressurized water reactor (PWR) 17x17 assemblies and supplemented earlier results obtained for boiling water reactor (BWR) assemblies. Code validations based on PWR and BWR experimental results considerably enhance severe accident code capabilities. This project to evaluate BWR expermiental work been completed	SNL			17	NUREG/CR-7143
①	REVEKA	Clad Ballooning The general objectives of ballooning experiments are to measure the amount of cladding strain and coolant channel blockage which occurs when cladding deforms at high temperatures (over 1000K) under loss-of-coolant accident (LOCA) conditions. REBEKA tests 5, 6 and 7: 49 rods, electrically heated over length 3.9m, internal/system pressures 6.00/0.45MPa, initial heatup 7K/s, burst temperatures 1048-1073, 1038-1063 and 1028-1063K, burst strain ranges (mean) 39-88(49), 32-64(42) and 42-87(55)%	独　FZ Karlsruhe			9,18	

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②	SANDIA XR	溶融金属使用によるブロッケージ生成とリロケーション挙動試験 The XR (Ex-Reactor) tests, conducted at Sandia National Laboratories in the period 1993-96 for the U.S. Nuclear Regulatory Commission, are out-of-pile experiments focused on core melt progression behaviour in boiling water reactors, with an emphasis on behaviour of relocating metallic melts. Two preliminary experiments and one large-scale experiment were performed.A test section including structural details of the lower BWR core region is preheated in an argon-inert environment so that an accident-typical axial thermal gradient exists. Test sections include BWR control blade, channel boxes and fuel rods (XR2-1 pending) in addition to structural features such as fuel canister nose pieces and lower core plate. Metallic melts corresponding to control blade alloy and Zircaloy core components are prepared and introduced to the test section, simulating the melting and draining of core metallic components into the lower metre of the core	米国SNL	1993		1,9,18	
②	VEGA	照射済燃料からのFP放出挙動 VEGA(Verification Experiments of radionuclide Gas/Aerosol release) program8) was conducted at Japan Atomic Energy Agency (JAEA) from 1999 to 2005. The program especially focused on the behavior of low-volatile radionuclide and actinides released from high burn-up UO2 or MOX fuel under high temperature above the fuel melting point and elevated pressure up to 1.0MPa.	JAEA	1999		19	
②	NSRR	未照射燃料及び照射済燃料を用いた事故時燃料挙動	JAEA				

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②	QUENCH	模擬燃料集合体の再冠水時の酸化、溶融挙動 The general objective of the QUENCH programme at Forschungszentrum Karlsruhe (FZKA,previously KfK) is to provide an extensive experimental database on quench of an overheated LWR core while in a mainly rod-like state, giving improved understanding of the effects of water addition at different stages of the rod degradation. More detailed aims are the determination of a failure criterion for oxidised cladding and of the hydrogen source term. The bundle experiments summarised here are supplemented by an extensive series of single-rod quench experiments and by measurements of hydrogen absorption and release by Zircaloy cladding. The main parameters so far considered are heat-up rate, degree of cladding preoxidation, flooding rate and temperature at the onset of quenching.	独KIT	1997		3、9、18	
②	VERDON	燃料ペレットからのFP放出	仏IRSN			5	
②	PHEBUS FP	燃料を原子炉で核加熱させた、溶融挙動、エアロゾル、ヨウ素挙動試験 Investigate in-pile the late phase of core degradation and FP release (volatile and non-volatile). Investigate in-pile the late phase of core degradation and FP (low volatile) and transuranic nucleides release Investigate in-pile the late phase of core degradation and FP (volatile and non-volatile) with carbon compounds release	仏IRSN,国際共同研究（JNES参加）仏 (IPSN/CEA - Cadarache Beginning end 1993 (first test FPT0). The sixth and last test is foreseen in 2005.	1993		1、9	
②	ACRR-MP	後期溶融進展 Fuel damage tests conducted in the Annular Core Research Reactor To investigate late phase melt progression in degraded fuel geometry, this included melting dynamics, molten pool formation, and growth in a debris medium, crust formation and failure and pool migration through blockages and intact rod structure	米SNL 1982 - 1989	1982		5、9,18	
②	ACRR-DC	Dry debris bed coolability and melt dynamics tests conducted in the Annular Core Research Reacto r To heat up and melt down a dry core debris bed to determine the heat transfer characteristics of a debris medium and to study the melt formation in a debris medium	米SNL 1982 - 1989	1982		9、18	
②	HEVA/VERCORS		仏			9	

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②	HI/VI	NUREG/CR-6261	米ORNL			9	
②	CRL		カナダ			9	
②	SASCHA		独			9	
②	SAFEST	SAFEST experimental facilities are unique in providing the possibility to perform experiments in specific fields of research on corium behaviour in severe accidents in main types of light water reactors including BWRs. Integrating European severe accident research facilities into a pan-European laboratory for study of corium behaviour in severe accidents.	FP7 Coordinated by KIT, the SAFEST consortium is small and consists of 5 research centres (CEA, MTA EK, JRC-ITU, SCK-CEN and UJV), 2 universities (KIT and KTH), and one industrial partner			10, 11	
②	ALISA	Large-scale facilities of the ALISA project are designed to resolve the most important remaining severe accident safety issues, ranked with high or medium priority by the SARP group for SARNET NoE. These issues are coolability of a degraded core, corium coolability in the RPV, possible melt dispersion to the reactor cavity, molten corium concrete interaction and hydrogen mixing and combustion in the containment.	FP7 Coordinated by KIT			10, 12	

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②	TMI-2	As a consequence of the TMI-2 accident, the nuclear community embarked on a thorough review of the causes and consequences of severe core damage accidents. The OECD has sponsored an ambitious international cooperation programme to evaluate and understand the TMI-2 accident. This was a three-tier programme which included a computer code benchmark exercise (1989), post-irradiation examination of samples from various parts of the core (1990), and the recently concluded TMI-2 Vessel Inspection Programme (1993). The TMI-2 analysis programme has allowed reasonable estimation of what happened during the accident, and has provided very valuable data. Though not a planned experiment, where boundary conditions are well defined and dedicated data gathering systems are used, the TMI-2 accident provides the only severe accident data base for an integrated, full scale, PWR, available for benchmarking computer codes and for comparing with results obtained in small-scale experimental facilities		1979		18	
②	SCARABEE BF	The SCARABEE BF test series was performed by IPSN/CEA in the SCARABEE reactor at Cadarache, France, in the period 1985 to 1989. It was aimed at studying in the framework of fast reactor safety analysis the behaviour of a volumetrically heated molten fuel pool resulting from a sub-assembly melting at full power. The first test BF1, with pure molten UO2, has a rather fundamental character interesting also for other accident situations, including PWR severe accident onditions,characterized by the existence of a molten pool. This configuration is of interest for studying the heat flux distribution at the boundary of the pool and the corresponding heat transfer coefficient.	仏CEA	1985		9、18	
③	RASPLAV/MASCA	U02混合物を用いた圧力容器下部ヘッド内デブリ挙動と外部冠水冷却	OECD計画（NUPEC参加）			1	
③	圧力容器内デブリ冷却試験	U02混合物を用いた圧力容器下部ヘッド内デブリ冷却挙動	NUPEC			1	
③	ALPHA	アルミナを用いた圧力容器下部ヘッド内デブリ冷却挙動	原研			1	
③	BWRSAT計画	BWR下部ヘッドデブリの構造と特性を調べる試験 BWRSARコードへの反映	米国ORNL			5	
③	SONATA	アルミナ又は鉄との混合物を用いた圧力容器下部ヘッド内デブリ冷却挙動	韓国KAERI			1	

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③	SSWICS	圧力容器下部の水中へ落下したコリウムの挙動	米国ANL			4	
③	LIVE	2次元及び3次元実験装置での圧力容器下部の溶融デブリの挙動 層分離、熱伝播	独KIT			3	
③	PHEBUS-FP	燃料を原子炉で核加熱させた、溶融挙動、エアロゾル、ヨウ素挙動を把握（大規模総合試験） The objective is to investigate the main phenomena governing degradation of fuel as well as Fission Product (FP) release, transport and behaviour in the containment during a beyond design basis accident occurring in a Light Water Reactor. The scope involves release of FPs from degraded irradiated fuel as well as FP/aerosol physics and chemistry both in the primary circuit and in the containment building. The bundle aspect involves the late phase of core degradation up to the fuel melting with the related FP release (volatiles and non-volatiles). These tests will enable investigation of specific phenomena not observed in the previous PHEBUS SFD programme: simultaneous Zr oxidation escalation-melting-interaction with UO2 in an oxidising environment, effect of control rod materials (Ag-In-Cd or B4C), chemical interactions at high temperature, solid debris bed evolution up to molten pool formation, and UO2 oxidation by steam and air	仏IRSN,国際共同研究（JNES参加） The Phebus FP programme is led by the Institut de Protection de Sûreté Nucléaire (IPSN) of the French Commissariat à l'Energie Atomique (CEA) and the Commission of the European Communities (CEC) with international participation from USNRC, COG, NUPEC, JAERI,KAERI, HSK and PSI			1,18	
③	VEGA	実燃料による高温・高圧下での放射性物質放出挙動	原研			1	
③	CALO	下部ヘッドのコリウム流下挙動、冷却性評価試験	仏IRSN			5	
③	MELT-II	ナトリウム冷却高速炉のＦＣＩ 溶融燃料と燃料流出経路が隣接する配置を模擬した試験装置及び模擬物質を用いた可視化基礎試験	ＪＡＥＡ				
③	IVMR	It aims at providing new experimental data and a harmonized methodologyfor in-vessel melt retention (IVR). The IVR strategy for LWR intends to stabilize and isolate corium and fission products inside the reactor pressure vessel and in the primary circuit.	cordinated by IRSN NITI（露）、CERES(ハンガリー) UJV（チェコ）他			10,13	
③	QUESA	This project aims at studying and modelling more precisely the way the oxide layer is formed. It would be also an opportunity to know the influence of this kind of atmosphere on hydrogen production, for instance during bundle reflood.	EDF (France),5 partners (GRS, IRSN, LEI, PSI plus IBRAE in Russian Federation)			10,13,20	

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③	BALI	Melt Pool Thermal Hydraulics The general objectives of melt pool thermohydraulics experiments are to measure heat flux distributions in different geometries over a wide range of internal heat generation and to quantify them, usually in the form of correlations Nu = f(Ra) (Nusselt number as a function of Rayleigh number) for averaged and local heat transfer coefficients between melted fuel and crust or surroundings. The melt pool may be formed in the core region or in the lower plenum. Recent experiments explore solidification and crust formation too.	BALI, STR/CEN Grenoble, France			9,18	
③	RASPLAV	Melt Pool Thermal Hydraulics	露 Russian Research Institute "Kurchatov Institute", supported by OECD			9,18	
③	SIMECO	Melt Pool Thermal Hydraulics	スウェーデン Royal Institut of Technology (RIT)			9,18	
③	BESON	Gap Thermal Hydraulics The general objectives of gap thermal hydraulics experiments are to explore the cooling potential of a gap between core debris (solid or molten) and RPV wall. Further application is the extrapolation to in-vessel core retention and coolability for present and future reactor design (core catcher)	独 Siemens, Erlangen, and supported by BMWi, Germany			9,18	
③	CORCOM	Gap Thermal Hydraulics	独Technical University Munich and supported by BMWi			9,18	
④	LHF	下部模擬ヘッドを高温下加圧したクリープ挙動	米国SNL			1	
④	OECD-LHF	上記LHF試験のOECD/NEA国際協力プロジェクト	OECD計画			1	

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④	FOREVER	模擬下部ヘッドを高温下加圧したクリープ挙動（多数の小規模分離効果試験） Investigation of the gap formation between the debris and the lower head vessel wall and evaluation for the effect of the gap on the cooling characteristics of the lower head vessel.	スウェーデン王立工科大学			1,9,18	
④	COTELS	UO2混合物を用いた圧力容器外条件下での溶融物/冷却材相互作用（FCI)試験	OECD計画（NUPEC参加）			1	
④	ALPHA	アルミナ又はステンレス鋼を用いたFCI試験	原研			1	
④	FARO	UO2混合物を用いた主に高圧条件下でのFCI試験 The FARO test programme has been designed to improve understanding on molten core/coolant/structure interaction by performing experiments at large scale in realistic conditions in order to reduce the uncertainties on the reactor system response to melt progression and relocation. In the FARO facility, the penetration and quenching of molten corium into the water of the lower plenum or reactor cavity and its subsequent settling on the bottom structures are simulated. The conditions are realistic for the melt composition (UO2-ZrO2-Zr), for the water depth (up to 2 m) and for the system pressure and temperature (up to 5.0 MPa, 536 K). Melt quantities up to 200 kg and 3180 K are used.	伊イスプラ研究所	1991		1,9､18	
④	KROTOS	UO2混合物又はアルミナを用いた主に低圧条件下でのFCI試験 The main objective of the KROTOS tests is to investigate steam explosions and, in particular, different stages of it: pre-mixing, triggering, propagation and expansion which determine the energetics and structural loading. Typically, these experiments are designed to help directly modelling efforts of the separate phases of energetic steam explosions by maintaining welldefined test geometry and conditions. KROTOS tests have been performed with various simulant materials, such as tin and alumina, or with a prototypic corium mixture (80 wt% UO2+ 20 wt% ZrO2) with masses ranging from 1.5 kg to 6 kg.	伊イスプラ研究所	1991		1,9､18	
④	TROI	TROI is a facility dedicated to steam explosion experiments located in KAERI (South Korea).	韓国KAERI			19	
④	STORM	エアロゾルの配管内での沈着・再浮遊挙動	EU/JRC			1	
④	WIND	エアロゾルの高温・高圧配管内での沈着・再浮遊挙動	原研			1	

研究項目分類	プロジェクト名	試験内容	実施機関	年開始	重要度	出典	報告書等
④	TRANSAT/TUBA/DEVAP	一次系配管(ホットレグ、蒸気発生器伝熱管)内のエアロゾルの沈着、ガス状放射性物質の化学吸着	仏IRSN			1	
④	CHIP	PWR一次冷却系でのヨウ素蒸発、Cs挙動(Phebusu知見の反映)	仏IRSN			5	
④	LAVA	Investigation of the gap formation between the debris and the lower head vessel wall and evaluation for the effect of the gap on the cooling characteristics of the lower head vessel.	韓 Korea Atomic Energy Research Institut (KAERI)			9,18	
⑤	WC	酸化鉄/アルミナを用いたZion炉型キャビティ模擬の1/10スケール試験	米国SNL			1	
⑤	COREXIT	実炉溶融物を用いたZion炉型キャビティ模擬の1/40スケール試験	米国SNL			1	
⑤	SERTCY	酸化鉄/アルミナを用いたZion炉型キャビティ模擬の1/20スケール試験	米国SNL			1	
⑤	LACOMECO	模擬コリウムと冷却水の反応	OECD計画　独KIT			2	
⑤	SCV試験	国内MARKⅡ改良型PCVの1/10縮尺モデル加圧試験	NUPEC			1	
⑤	PCCV試験	PCCVの1/4縮尺モデル加圧試験	NUPEC/米国NRC			1	
⑤	RCCV試験	基礎要素試験	NUPEC			1	
⑤	RCCV試験(米国)	RCCVの1/6縮尺モデル加圧試験	米国SNL			1	
⑤	PCCV試験(英国)	Sizewell-Bの1/10縮尺モデル水圧加圧試験	英国			1	
⑤	Hydrogen Combustion in the course of DCH	縮尺1/18, 1/7モデルをDCH時の格納容器内での水素ガス燃焼実験	独KIT			6	
⑥	COTELS	UO2混合物を用いた誘導加熱による2次元コンクリートトラップを使用した試験	NUPEC			1	
⑥	SWISS	ステンレス鋼を模擬デブリとした誘導加熱試験	米国SNL			1	
⑥	WTECOR	酸化アルミナを模擬デブリとした直接通電試験	米国SNL			1	
⑥	MACE	UO2酸化物を用いた直接通電試験	米国ANL			1	
⑥	DISCO-H	PWRの格納容器内でのFCI挙動	独KIT			3	
⑥	HDR					7	

研究項目分類	プロジェクト名	試験内容	実施機関	年開始	重要度	出典	報告書等
⑥	SERENA	炉外FCIに関連する主要な不確実性の定量化を目的。特に溶融物の過熱度、初期圧力、溶融物組成、空間幾何形状効果を含む水蒸気爆発への影響に関する知見を拡充することを目的に実験を実施。 SERENA is an OECD programme on fuel-coolant interaction (FCI), which has the scope of making a status of the code capabilities to predict FCI induced dynamic loading of the reactor structures (Phase 1), and performing the complementary research possibly needed to increase the level of confidence of the predictions (Phase 2). Phase 1 has been completed. It consisted of comparative calculations by available tools of selected existing experiments and reactor cases, in order to identify those areas where lack of understanding induced large uncertainties in the predictions of the loads in reactors1. Phase 2 has the scope of carrying out the confirmatory analytical and experimental research needed to reduce these uncertainties to acceptable level for risk assessment. Phase 1 was the first comparative exercise undertaken since ISP-39, which however concerned premixing	OECD/NEA			19	
⑦	OECD-MCCI	UO2酸化物を用いた2次元コンクリート浸食試験	OECD計画（NUPEC参加）			1	
⑦	MOCKA	Small-scale tests on MCCI Large-scale tests on MCCI	OECD計画　独KIT			2	
⑦	CCI	Coriumu/Concrete Interaction Experiment OECD MCCIプロジェクトの一環として米国ANLで実施された。コンクリート侵食が進んだ状態で注水した場合の溶融物の挙動調査を目的としたもの。	米国ANL			4	
⑦	MACE	Melt Attack and Coolability Experiments (MACE), 原子炉容器外のコリウム冷却とＭＣＣＩ	米国ＥＰＲＩ			7	
⑦	ACE	MCCIにおける熱水力学的及び化学的プロセスを検証し関連コードのデータベースを拡充することを目的として、国際的に支援されたACE(Ａｄvanced Containment Experiments)プログラムの一環として米国ANLで実施された。	米国ANL			7	
⑦	SURC	米国SNLで行われたMCCI実験の一つ。本実験は国際標準問題（ISP-24)に選定されている。	米国SNL			7	
⑦	DEFOR	DEFOR(Debris Bed Formation)計画はスウェーデン王立工科大学で実施されており、種々の条件で水プールに模擬溶融物が投入された際の、溶融炉心の細粒化試験。	スウェーデン王立工科大学			7	

研究項目分類	プロジェクト名	試験内容	実施機関	年開始	重要度	出典	報告書等
⑦	FARO	欧州JRCのイスプラ研究所における実験。圧力容器内を対象に溶融物が水プールに落下した場合の水蒸気爆発の発生を目的として高圧条件での実験が行われてきたが、圧力容器外を対象とした実験も実施。 In addition, two experiments have been performed related to testing the performance of ex-vessel core catcher based on	伊イスプラ研究所			7,18	
⑦	WETCOR	直接通電加熱したAl203等の模擬溶融物と石灰岩系コンクリートとの反応中に注水し、溶融物の冷却性を調べた実験。	米国SNL			7	
⑦	ＰＵＬｉＭＵＳ	浅い水プールに模擬溶融物を流入させ、その広がり挙動を調査。	スウェーデンＫＴＨ			7	
⑦	COMET		独			9	
⑦	DEBRIS		独			9	
⑦	POMECO		スウェーデン			9	
⑦	STYX		フィンランド			9	
⑦	OECD/NEA- MCCI-Ｉ	The focus of the original project, MCCI-I (completed), was to investigate the coolability of molten core materials, interacting with the containment structural concrete, by an overlaying water layer	OECD/NEA, Edf-IRSN/CEA, NRC-ANL			16	
⑦	OECD/NEA- MCCI-Ⅱ	National Laboratory (ANL), a second project using the same ANL facilities was set-up. The MCCI-2 program was carried out from 2006 to early 2010 to help bridge data gaps not fully covered in MCCI-1. Testing falls into four categories: ·Combined effect tests to investigate the interplay of different cooling mechanisms and to provide data for model development and code assessment. ·Tests to investigate the effectiveness ofnew design features that enhance debris coolability. ·Tests to generate additional 2-D core-concrete interaction data for model development ·and code validation. ·Integral test at larger scale to confirm synergistic effect of different cooling mechanisms and to provide data for validation of severe accident codes. Aside from these tests, a supporting analysis task was carried out to further develop/validate debris coolability models that form the basis for extrapolating the experiment findings to plant conditions. In total, 10 tests were conducted in this program (all successful).	OECD/NEA, Edf-IRSN/CEA, NRC-ANL	2006		17	

研究項目分類	プロジェクト名	試験内容	実施機関	年開始	重要度	出典	報告書等
⑧	BMC	区画間の可燃性ガスの挙動を把握	独GRS			1	
⑧	可燃性ガス濃度分布・混合挙動試験	格納容器を模擬した多区画容器内での挙動を把握	NUPEC			1	
⑧	可燃性ガス燃焼挙動試験(大規模)	格納容器を模擬した多区画容器内での挙動を把握	NUPEC			1	
⑧	HYMERES Hydrogen Mitigation Experiments for Reactor Safety	格納容器内での水素挙動を緩和対策（スプレイ、冷却装置、PAR等）を含めて PANDA (PSI) と MISTRA (CEA)の実験装置を用いて実施	OECD計画 スイスPSI,仏CEA			2	
⑧	PITEAS	エアロゾルの格納容器内での粒子成長・沈着挙動	仏IRSN			1	
⑧	放射性物質除去効果試験	エアロゾルの格納容器スプレイによる除去効果	NUPEC			1	
⑧	放射性物質捕集特性試験	エアロゾルの格納容器貫通部での補修効果	NUPEC			1	
⑧	RTF	放射線場での格納容器内ガス状ヨウ素挙動	加AECL			1	
⑧	CAIMAN/IODE	放射線場での格納容器内ガス状ヨウ素生成移行挙動	仏IRSN			1	
⑧	スプレイ除去効果試験	ガス状ヨウ素の格納容器スプレイによる除去効果	原研/産業界			1	
⑧	ARTIST	蒸気発生器伝熱管破損時（シビアアクシデント条件）のエアロゾル・液滴挙動とシビアアクシデントマネジメント効果	スイスPSI国際共同研究（JNES参加）			1	
⑧	STEM	格納容器内及び1次冷却系配管内でのヨウ素、ルテニウムの化学的反応を考慮したソースターム評価	仏IRSN This project, led by IRSN and conducted at the IRSN facilities (EPICUR, START) in Cadarache (France) will be completed in 2015.			2	

研究項目分類	プロジェクト名	試験内容	実施機関	年開始	重要度	出典	報告書等
⑧	BIP-2	ヨウ素と格納容器表面の塗料との吸着・離脱、有機ヨウ素の生成挙動	加AECL The three-year follow-up project, led by AECL, started in April 2011 and the final report is to be issued in 2015.	2011		2	
⑧	THAI	格納容器内での水素、ヨウ素及びエアロゾルの挙動 PAR性能試験	OECD計画　独　Becker Technologies				
⑧	EPICUR	Co60照射施設、I-131トレーサーによるエアロゾル挙動	仏IRSN			5	
⑧	ERCOSAM	格納容器内の多様な混合現象による水素ガス挙動 オランダのBrossele発電所（PWR)の格納容器設計をベース	独KIT			6	
⑧	TOSQAN	The Ercosam European project This FP7 European project is studying in particular the phenomenon of de-stratification of hydrogen induced by spraying at various experimental scales. These studies complete the results of the TOSQAN programme carried out between 2000 and 2008. heat and mass transfer in a spray for containment application	仏ＩＲＳＮ	2000		9	
⑧	PANDA	格納容器内でのガスの混合、層化現象	スイスPSI			9	
⑧	CONTAIN	the thermal hydraulics in the NUPEC 1/4-scale model containment experiments	ＮＵＰＥＣ			9	
⑧	MISTRA		仏			9	
⑧	IPRESCA	This project has just been launched with more than 25 partners on the basis of their own resources in the TA2/SARNET. It aims to better integrate the international research activities related to the pool scrubbing phenomena by providing support in experimental research to broaden the current knowledge and database, and by supporting analytical research to facilitate systematic validation and model enhancement of the existing pool scrubbing codes.	Becker Technologies (Germany) and 25 partners (from Europe and beyond)			10,16	
⑨	LSVCTF	燃焼挙動に関するベントの影響等を把握	加AECL			1	
⑨	HYKA	過酷事故時の水素挙動と緩和対策の大規模実験	独KIT			3	

研究項目分類	プロジェクト名	試験内容	実施機関	年開始	重要度	出典	報告書等
⑫	原子炉格納容器信頼性実証試験	Mark-Ⅱ改良型鋼製格納容器を模擬した1／10 スケールの試験体が破損するまで加圧する試験を実施した結果，約4.6MPa[gage]まで破損が生じない 電気配線貫通部モジュールを対象として，200℃，0.8MPa における気密性の確認と漏えいが発生する温度・圧力条件の確認試験 （H15.3総括報告書）	NUPEC			8	
⑫	格納容器電気ペネトレーションの特性試験	実機の電気配線貫通部の構造を反映した試験体を用い，ＬＯＣＡ時の圧力，温度条件を超える条件下で気密性能について検証（昭和62年度）	電力共研			8	
⑬（ア）	CYBL	Ex-vessel Thermal Hydraulics Investigation of ex-vessel cooling as accident management measure, to enlarge the data base on critical heat flux for model development and to verify the cooling capability under reactor typical condions	米 Sandia national Laboratory (SNL) and supported by USNRC			9,18	
⑬（イ）	NTS	自由空間における燃焼挙動を把握	米国DOE Nevada Test Site			1	
⑬（イ）	RUT	爆轟現象の把握	露クルチャトフ研究所			1	
⑬（イ）	ENACCEF		仏IRSN			9	
⑬（イ）	CTF		カナダ			9	
⑭（イ）	VULCANO	５０ｋｇ程度のウランを含む溶融物によるMCCI試験 コアキャッチャー試験	仏IRSN			5	
⑭（ウ）	PASSAM	It was mainly of an R&D experimental nature and aimed at investigating phenomena that might enhance atmospheric source term mitigation in case of a severe accident in a Nuclear Power Plant (NPP), mainly through the use of Filtered Containment Venting Systems (FCVS)	IRSN, EDF and University of Lorraine (France); CIEMAT and CSIC (Spain); PSI (Switzerland); RSE (Italy); VTT (Finland) and AREVA			10,14	
⑭（エ）	KALI	水素再結合、放出試験	仏IRSN			5	
⑭（エ）	触媒式FCS再結合装置適用性	PARの水素処理性能確認(平成１１年～平成１３年)	電力共研			8	
⑭（エ）	PAR試験（SNL）	PAR起動時の水蒸気の影響	米国SNL			8	
⑭（エ）	PAR試験（THAI）	格納容器内での水素、ヨウ素及びエアロゾルの挙動 PAR性能試験	OECD計画 独 Becker Technologies			8	

研究項目分類	プロジェクト名	試験内容	実施機関	年開始	重要度	出典	報告書等
⑭（カ）	AIR-SFP	AIR-SFP project has two main objectives: ·To assess more precisely the applicability of SA codes to the calculation of transients in SFPs by carrying out a benchmark with different SA code, including a criticality risk assessment. ·To elaborate a roadmap for further R&D on SFP accidents.	IRSN / CIEMAT / ENEA / ENGIE / GRS / IJS / IVS / KIT / LEI/NRG / NUBIKI / PSI / REL / SSTC NRS / UJV			15	
⑮	ＭＥＬＣＯＲ	NUREG/CR-6119, Vol. 3, Rev. 0 SAND2001-0917P MELCOR Computer Code Manuals Vol. 3: Demonstration Problems Version 1.8.5 May 2001 Revised October 2000 Printed May 2001 Prepared by R. O. Gauntt, R. K. Cole,	米国SNL				
⑮	ＭＡＡＰ	Modular Accident Analysis Program Technical foundation of reactor safety - knowledge base for resolving severe accident issues - Revision 1 EPRI Report No. 1022186	米国ＥＰＲＩ				
⑮	ＡＳＴＥＣ	The ASTEC integral code for severe accident simulation Nuc. Tech.2009	仏				
⑮	SAMPSON	Severe accident analysis code SAMPSON improvements for IMPACT project JSME int. journal 2001	日				
⑮	ＴＨＡＬＥＳ２	THALES2コードは、軽水炉プラント全体を模擬し、シビアアクシデント時のプラント内の熱水力的な現象とFP挙動を一括して考慮可能な確率論的安全評価(PSA：Probabilistic SafetyAssessment)用の総合的なシビアアクシデント解析コードである。	日ＪＡＥＡ				
⑮	M-RELAP5	制御系、熱水力系、熱構造材、原子炉動特性等の計算機能を有し、原子炉の事故時の熱流動解析を行う上で汎用性の高い計算コードであり、米国エネルギー省(DOE)及びアイダホ国立研究所(INL)により開発されたRELAP5-3D コードを基に、三菱重工業（株）がPWR の小破断LOCA 解析に適用するため、10 CFR 50 Appendix K “ECCS valuation Models”にて要求される保守的なモデル（Moody 臨界流モデル等）を付加したコード。				7	
⑮	SPARKLE-2	M-RELAP5 の炉心部分を３次元炉心動特性に置き換えたものである。M-RELAP5 から炉心入口条件(温度、流量等)を3 次元熱流動コードMIDAC に受け渡し炉内の1 次冷却材温度等を計算する。3 次元炉心動特性コードCOSMO-K は、それを用いて中性子束変化を計算し、その時刻における炉心発熱量が得られ、これをM-RELAP5 に受け渡すことで、炉心内の詳細な核的フィードバックを考慮したプラント過渡解析を行う。				7	

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⑮	GOTHIC	原子炉格納容器システムの熱流動解析を主目的に開発された汎用解析コード。事故時の１次系からの冷却材流出、格納容器スプレイ応答、水素発生等を境界条件として、原子炉格納容器内の熱流動挙動を評価する。				7	
⑮	COCO	原子炉格納容器内を気相系と液相系に大別し、各系内では状態は一様とし、各々の系について質量及びエネルギー保存則を解く原子炉格納容器内圧解析コード。（設計基準事故の格納容器内圧解析に使用）				7	
⑮	JASMIN	水蒸気爆発現象の全過程を機構論的にシミュレートする水蒸気爆発解析コード（JAEA が開発し、実験再現解析等に使用				7	
⑮	SAFER	・長期間の原子炉内熱水力過渡変化及び炉心ヒートアップを解析するコードであり、原子炉圧力容器に接続する各種一次系配管の破断事故、原子炉冷却材流量の喪失事故及び原子炉冷却材保有量の異常な変化等を取り扱うことができる。BWRプラントの設計基準事故のLOCA解析に適用されている。				8	
⑮	CHASTE	・燃料集合体軸方向の任意の一断面で燃料ペレット、燃料被覆管、チャンネルボックス等の温度計算を行うコード。 ・燃料ペレットを半径方向に最大９ノードに分轄し、燃料集合体ない燃料棒を全て1本ごとに取り扱い、その熱的相互作用（輻射）も考慮している。 BWRプラントの設計基準事故のLOCA解析に適用されている。				8	