MAAPによる1～3号機の事故シーケンスの詳細解析について

Detailed analysis of the accident progression of Units 1 to 3 by using MAAP code

平成24年7月23日

東京電力株式会社

TOKYO ELECTRIC POWER COMPANY

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1．はじめに

- 平成23年5月、それまでに得られていた地震発生直後の状態や運転操作等に関する情報をもとに、MAAPコードを用いた福島第一原子力発電所1～3号機にかかる事象進展解析の結果を公表した。
- ただし、当時はプラント状態や運転操作に関する情報が十分に整理されていなかったことから、設計情報をそのまま解析条件として使用するなどした結果、解析結果と観測されるデータの間に齟齬がみられた。
- そのため、平成23年5月に解析結果を公表した以降も、運転員へのヒアリング、現場調査、プラントデータの収集を継続して実施してきた。
- 同時に、実機プラントデータや解析結果から、事故の推移を合理的に説明できるプラント状態の推定を継続して実施してきた。
- この解析は、現時点までに明らかになったプラントに関する情報とプラント状態に関する推定を元に、事故発生直後のプラント挙動をできる限り再現出来るように解析条件を設定し、解析を実施したものである。

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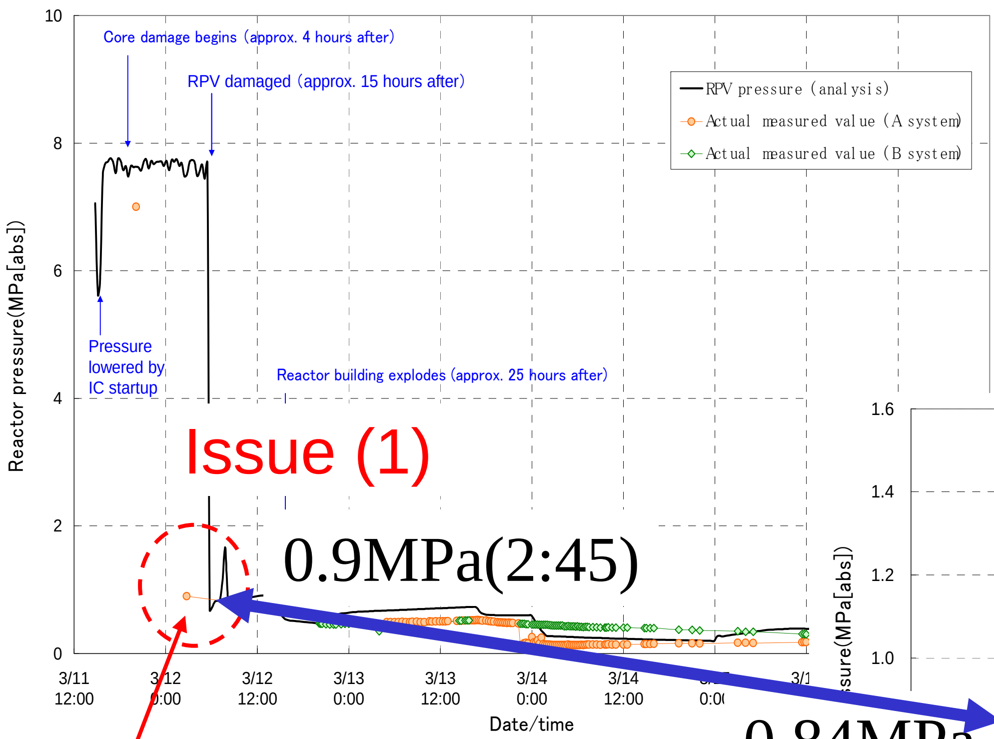
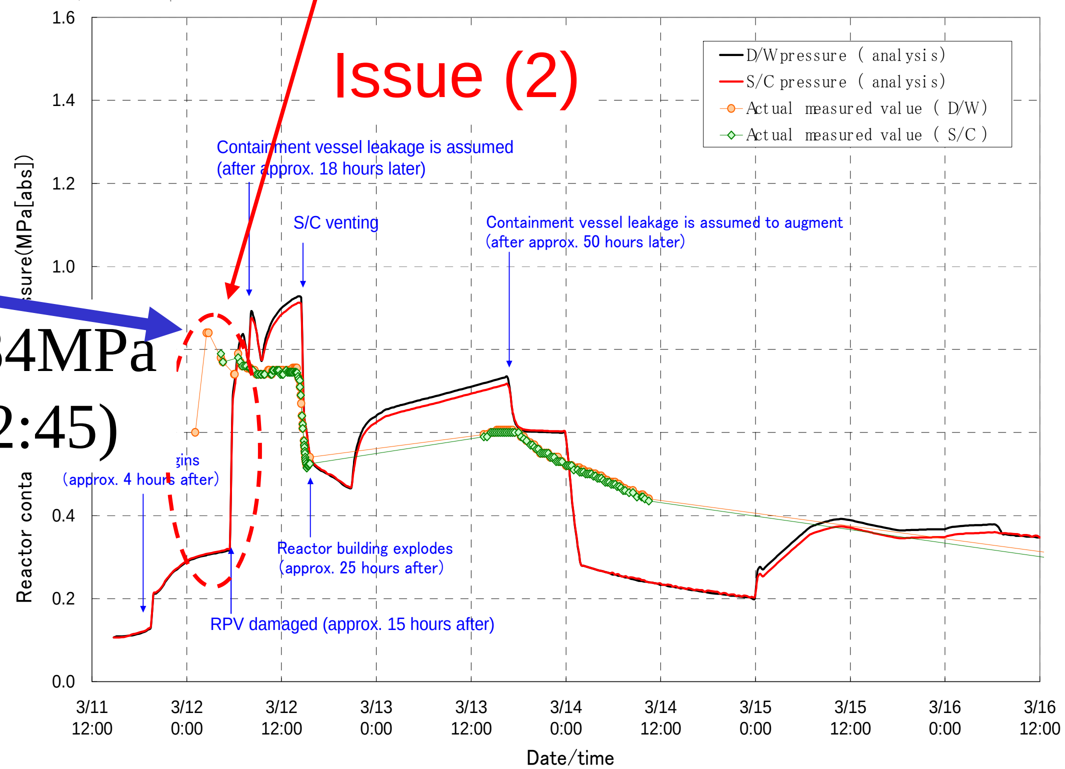
2．1．平成23年5月に公表した解析結果にみられた課題(1号機)



Calculated value of PCV pressure does NOT agree with measurement.
<u>measured > calculated</u>

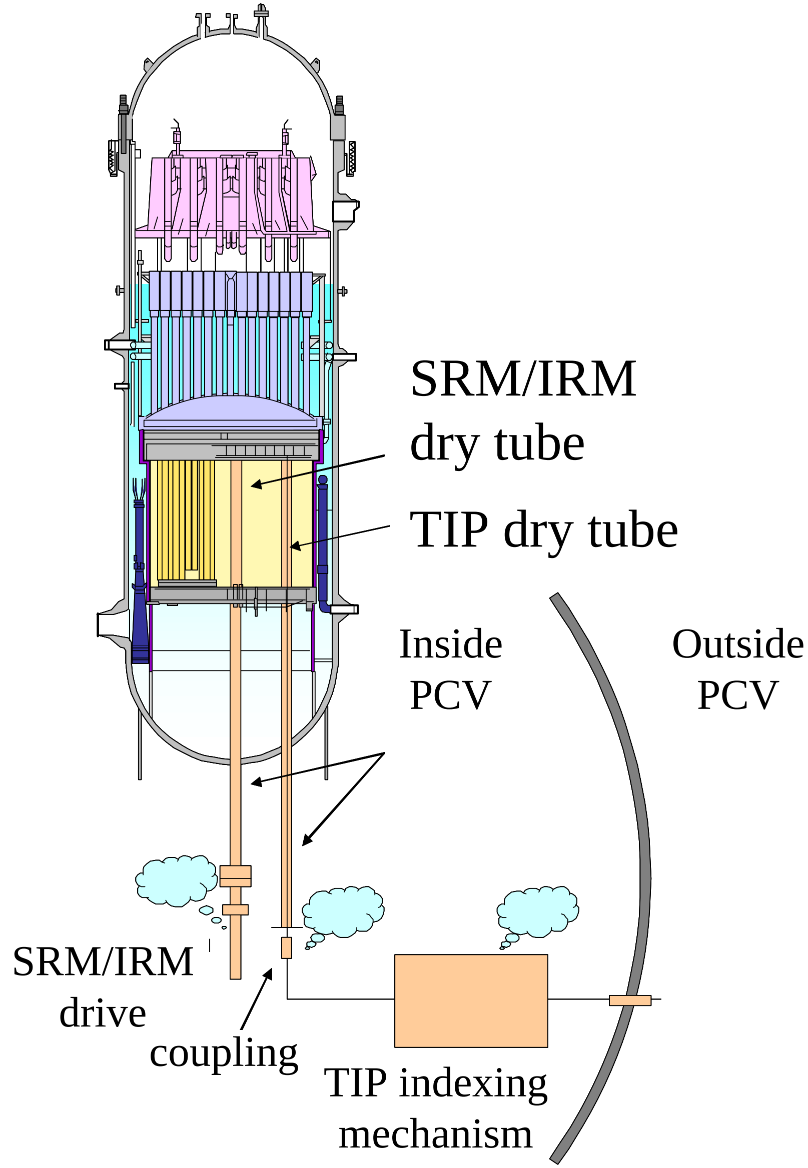


Calculated value of RPV pressure does NOT agree with measurement.
<u>measured < calculated</u>

2．2．気相部からの漏えい(1/2)

- ✓ When the water level is below TAF and the temperature of core region becomes high, there are some possibilities for failure of in-core monitors.
- ✓ Of these, SRM/IRM and TIP are installed in the dry tube which form part of the pressure boundary of the reactor pressure vessel.
- ✓ Therefore, failures of dry tubes result in gas leakage from RPV directly into D/W.

- In this analysis, we employed the analytical condition that the leakage will happen when PCT rise above 1000K.
- （cross section of leakage hole: 1.4cm²）



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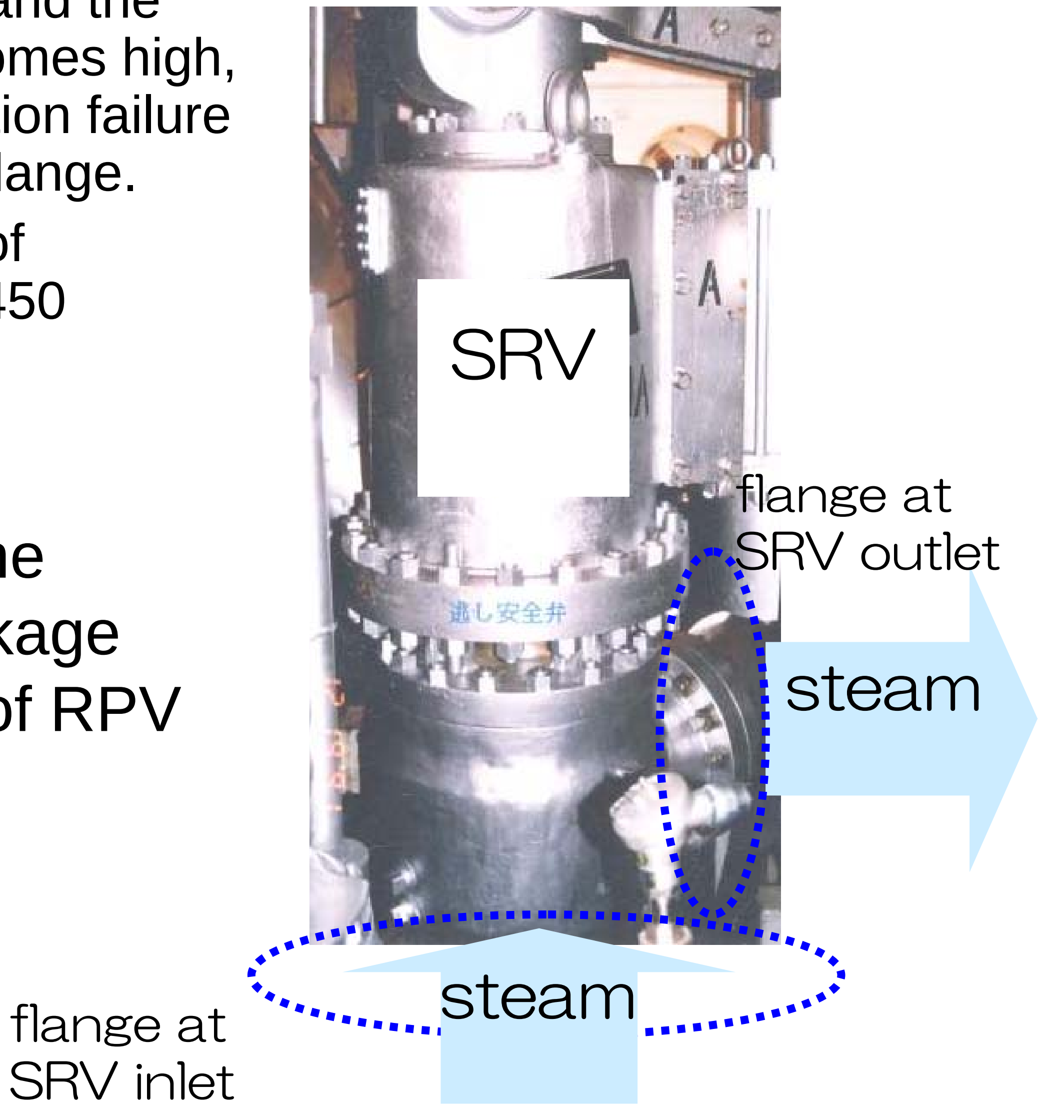
2. 2. 気相部からの漏えい(2/2)

- ✓ When the water level is below TAF and the gas temperature of core region becomes high, there are some possibilities for function failure of gasket seal used in the SV/SRV flange.
- ✓ Especially, upper temperature limit of expanded graphite gasket is about 450 Degree-C.

- In this analysis, we employed the analytical condition that the leakage will happen when temperature of RPV gas rise above 450 degree-C.
- （cross section of leakage hole : 13.6cm^2）



2．2．解析で採用した主な仮定（1号機）

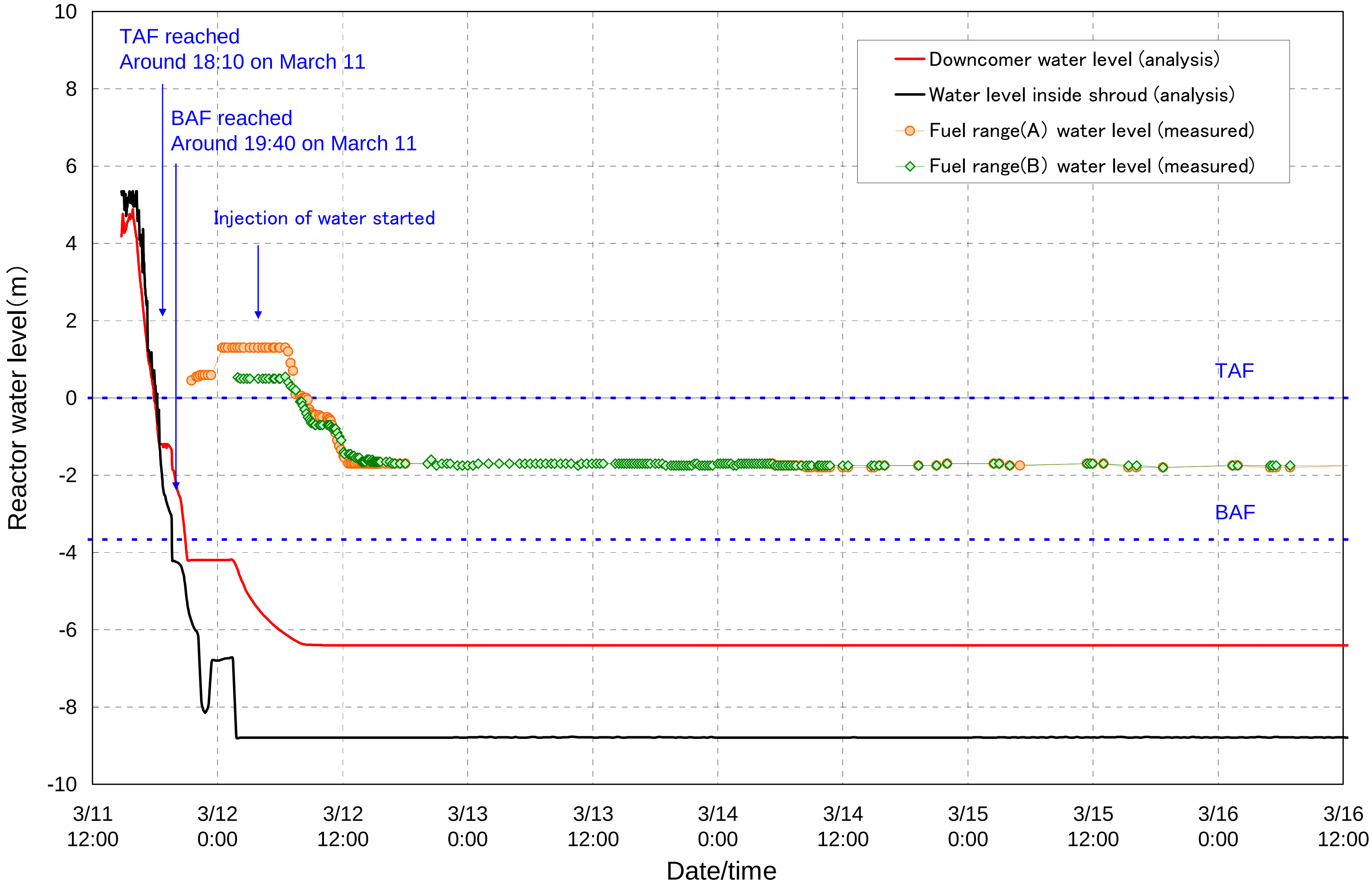
Assumptions

- After the Station Black Out due to Tsunami, Isolation Condenser (IC) does NOT worked.

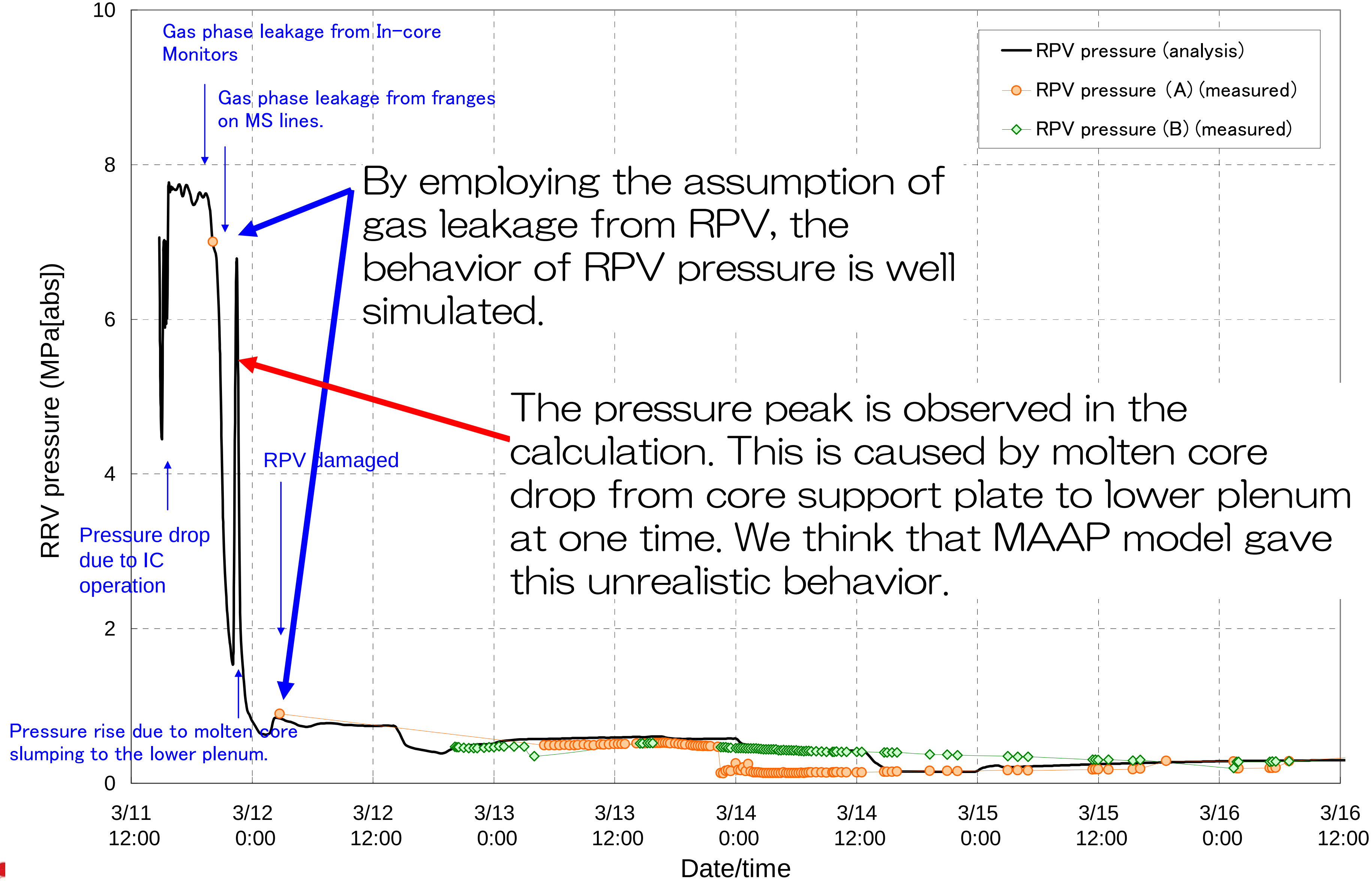
- The leakage from RPV will begin at the point that PCT rise above 1000K and temperature of RPV gas rise above 450 Degree-C, respectively. ⬅ Issue(1),(2)

- To simulate the behavior of PCV pressure decrease, the gas leakage from PCV will begin 12 hours, and be enlarged 50 hours and 70 hours after Earthquake, respectively.

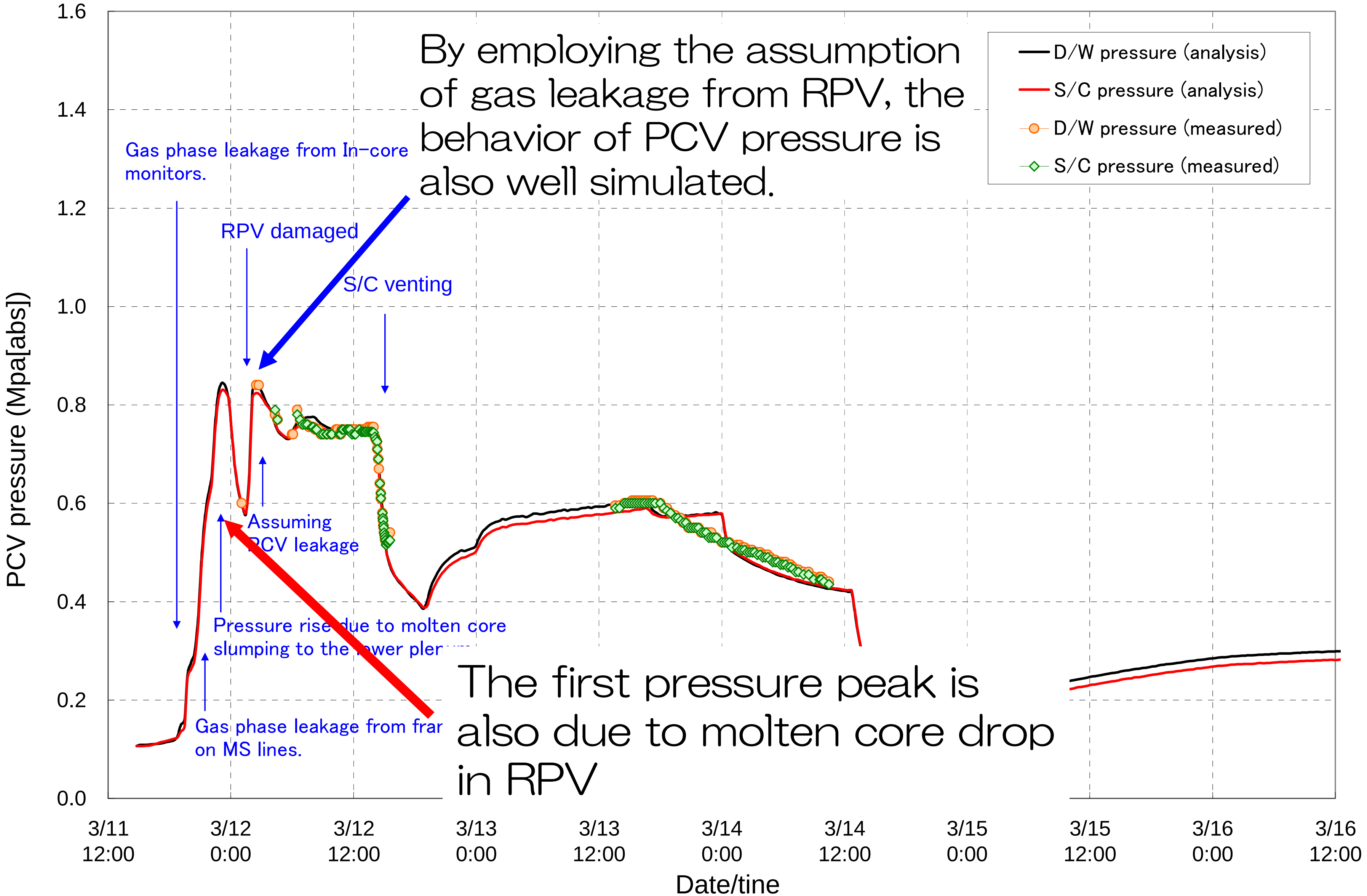
2．3．解析結果（原子炉水位：1号機）



2. 3. 解析結果（原子炉圧力:1号機）



2. 3. 解析結果（格納容器圧力:1号機）



2．3．解析結果のまとめ（1号機）

<Summary>

・Water level reached to TAF：3/11 18:10 （Approx. 3 hours after earthquake）

・Reactor core damage：3/11 18:50 （Approx. 4 hours after earthquake）

・RPV damage：3/12 1:50 （Approx. 11 hours after earthquake）

- As for issue 1 and 2: By employing the gas leakage from RPV directly into D/W, pressure of RPV and PCV are well simulated.
- Because of assumption for loss of function of IC after SBO, The reactor core was damaged in early stage. The RPV had damaged finally.
- Unrealistic pressure peak is observed in the calculation due to limitation of MAAP model.

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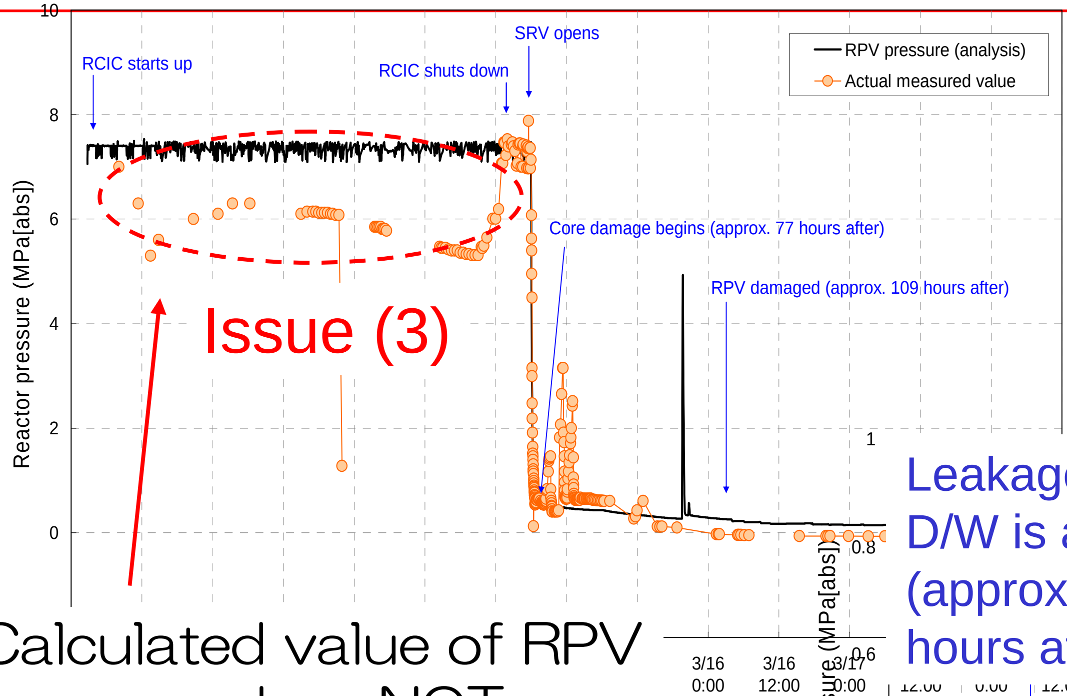
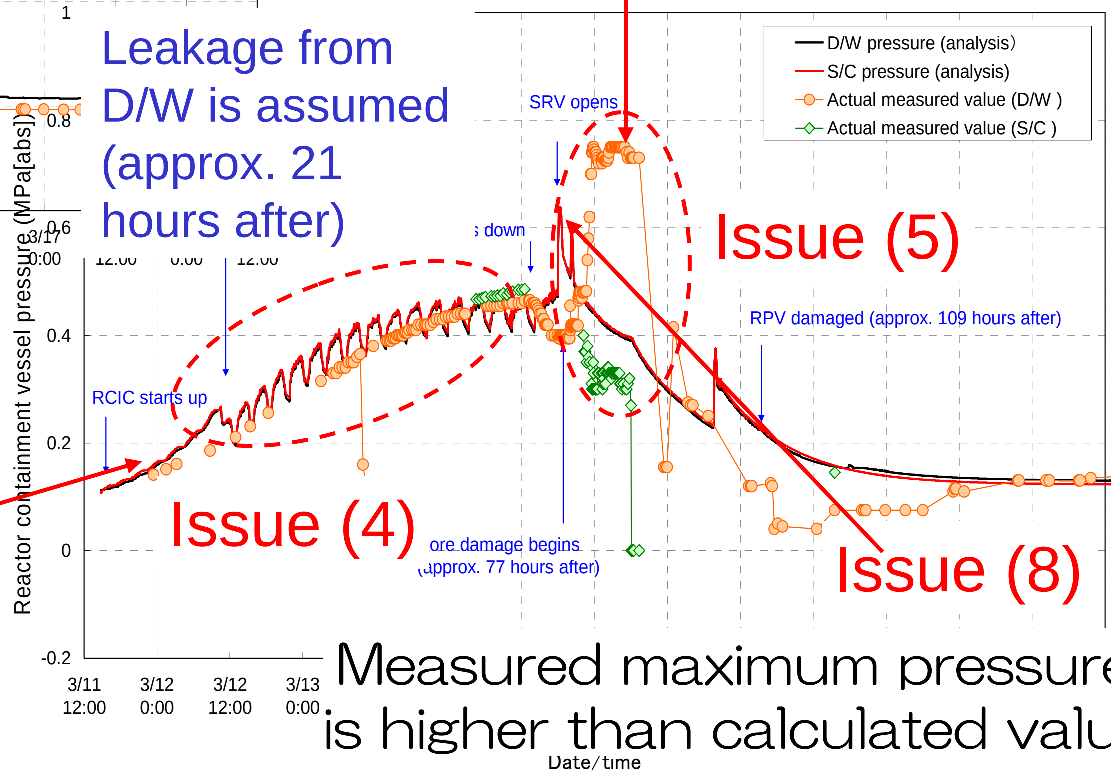
3. 1. 平成23年5月に公表した解析結果にみられた課題（2号機）



By employing assumption of gas leakage from PCV, Calculated value of PCV pressure decrease quickly, but measurement keep high pressure until morning on 3/15.

Calculated value of RPV pressure does NOT agree with measurement.



By employing assumption of gas leakage from PCV, calculated value of PCV pressure agrees with measurement.

Measured maximum pressure is higher than calculated value.

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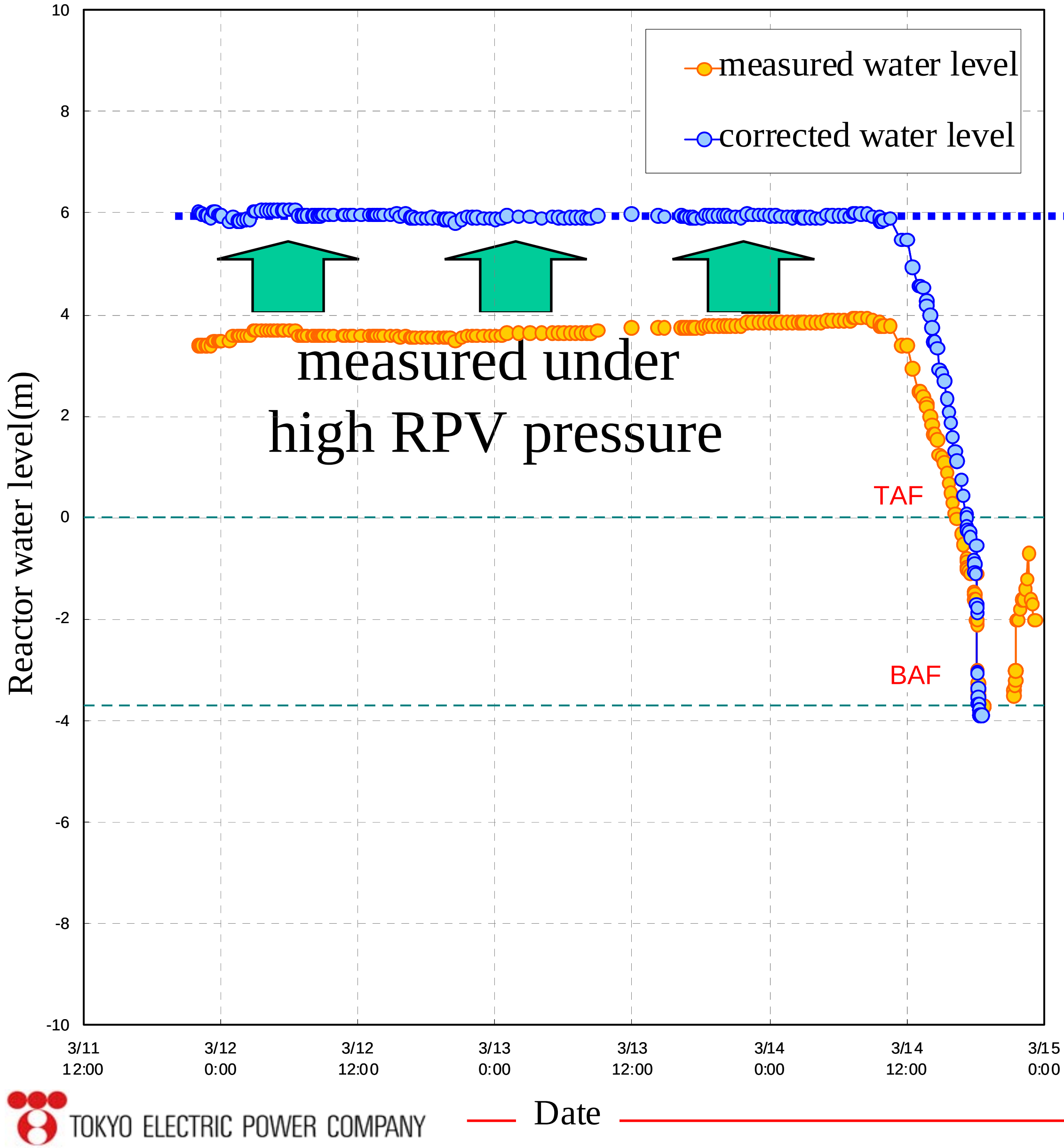
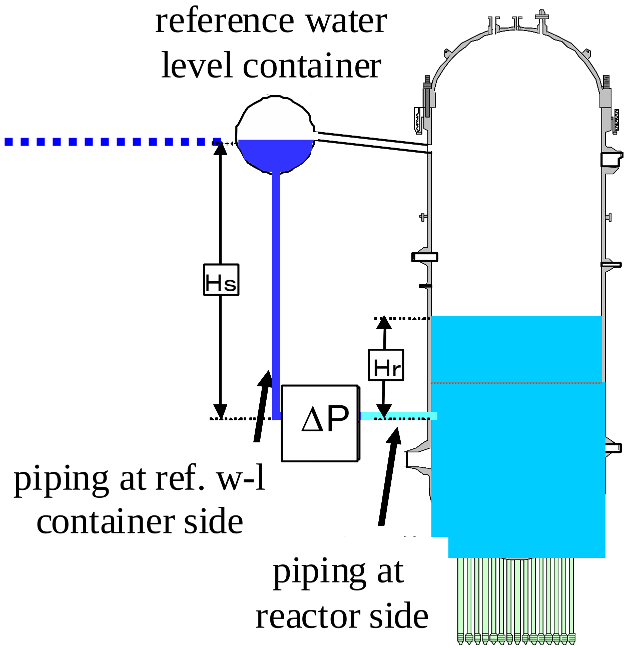
3．2．2号機のRCICの運転状態

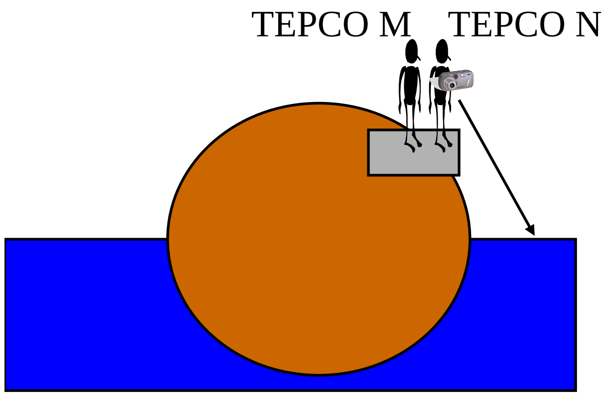




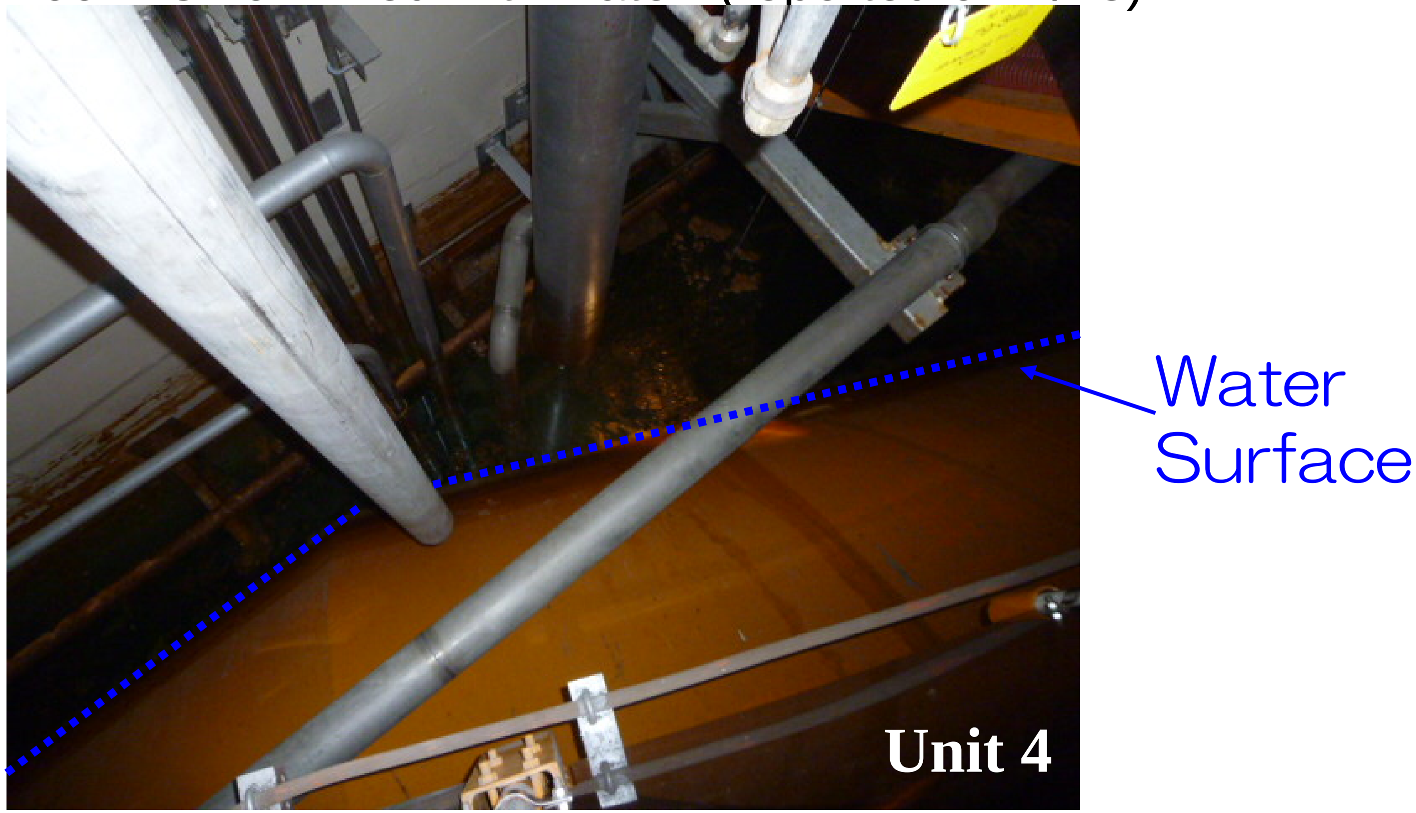
After correction, values of measured water level agree with maximum water level. Actual water level may be higher than that. Therefore, Two phase fluid will flow into RCIC turbine.

3．2．2号機のS/Cの状態

- The actual condition of torus room in unit 2 was not confirmed, but the torus room in unit 4 is filled with water. And about half height of the torus is submerged.
- If unit 2 was almost same situation, the heat stored in S/C would be transferred to the water.
- Also in unit 2, the torus room is now filled with water. (reported on 4/18)



Photograph from cat walk with the camera looking down



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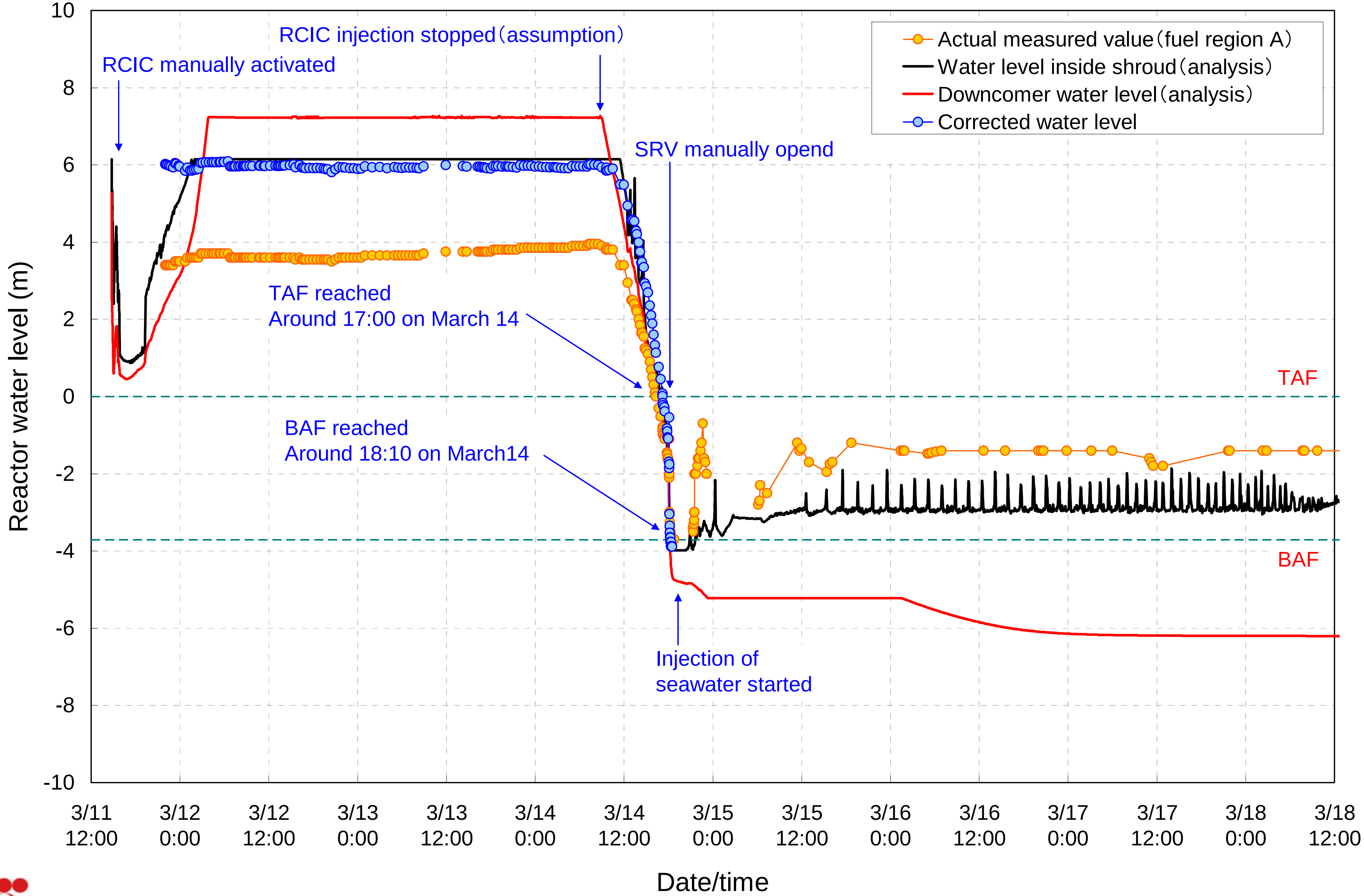
３．２．解析で採用した主な仮定（2号機）

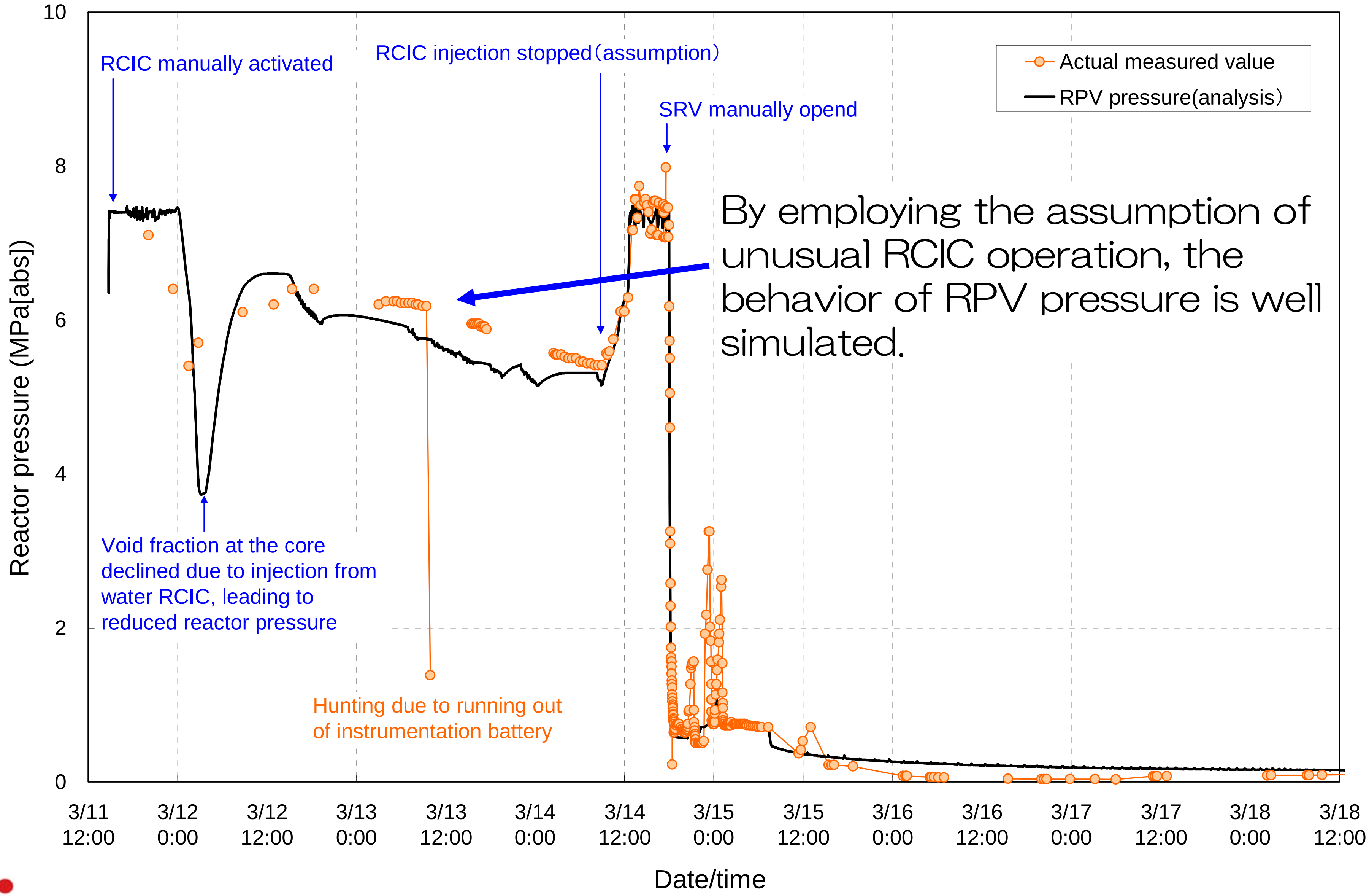
Assumptions

- To simulate the behavior of RPV pressure, which is less than the pressure of SRV activation, Two phase fluid will flow into RCIC turbine with thermal energy almost equal to decay heat. ← Issue(3)
- The heat will be removed from S/C surface by water in torus room. The torus room is gradually flooded by Tsunami seawater through penetration from turbine building. ← Issue(4),(5)
- The amount of injection water from fire trucks is set to keep water level around bottom of active fuel with considering hydrogen generation. ← Issue(8)
- To simulate the behavior of PCV pressure, Gas leakage from PCV will begin at 89 hours after earthquake.

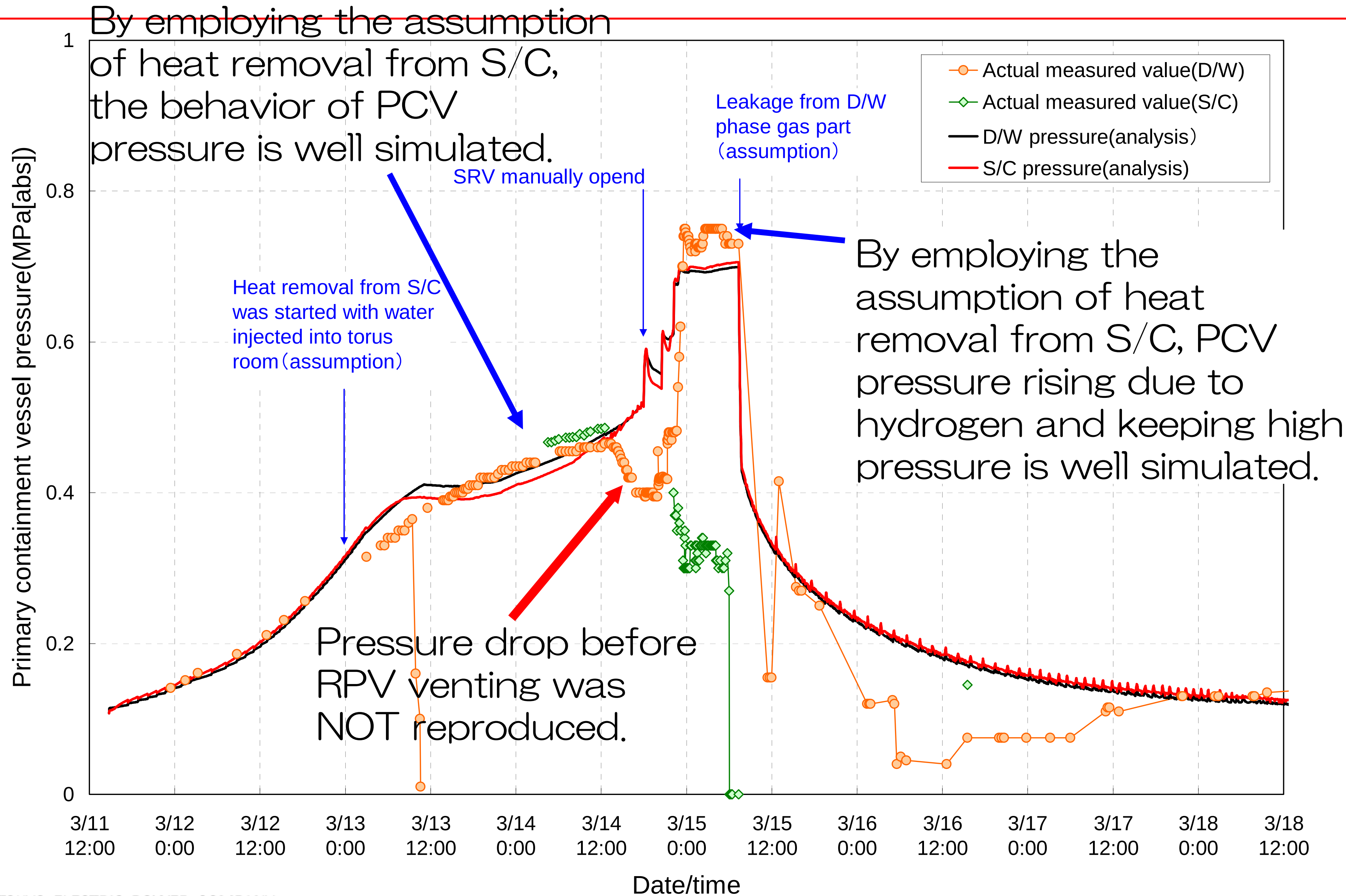
３．３. 解析結果（原子炉水位:2号機）



３．３. 解析結果（原子炉圧力:2号機）



３．３. 解析結果（格納容器圧力:2号機）



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3．3．解析結果のまとめ（2号機）

＜Summary＞

・Water level reaches to TAF：3/14 17:00 （Approx. 74 hours after earthquake）

・Reactor core damage：3/14 19:20 （Approx. 77 hours after earthquake）

・RPV damage：NOT damaged

- As for issue 3: The behavior of RPV pressure is well simulated by assumption of unusual RCIC operation.
- As for issue 4 and 5: The behavior of PCV pressure and maintaining high PCV pressure is well simulated by assumption of torus room flooding.
- As for issue 8: The maximum PCV pressure is well simulated by calibrating inject water from fire truck.
- Reactor core was damaged, but RPV was NOT damaged.
  (So many information derived from operation conducted in Fukushima site until today indicates a high likelihood of RPV damage.)

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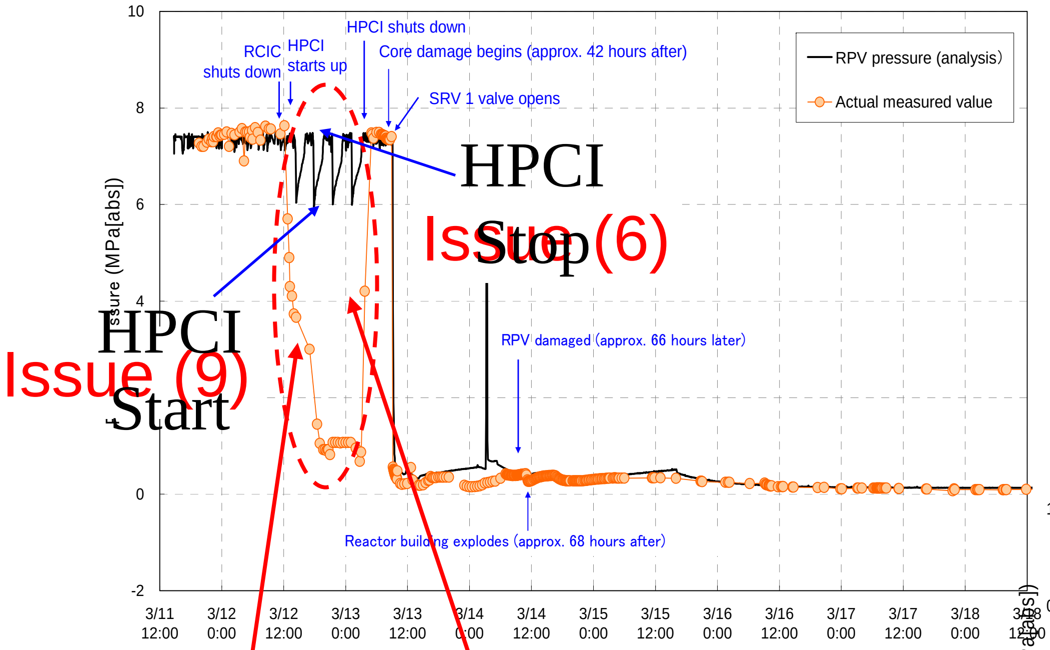
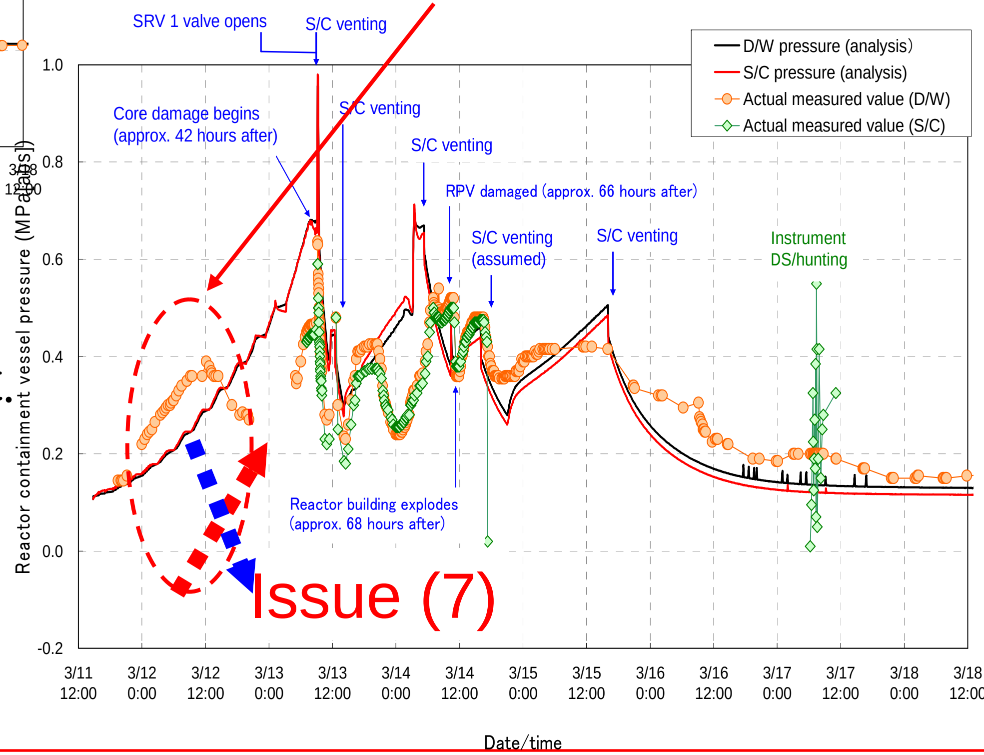
4．1．平成23年5月に公表した解析結果にみられた課題（3号機）



Calculated value of PCV pressure does NOT agree with measurement.
1: measured values are higher than calculated.
2: measured values are dropping but calculated values are rising.

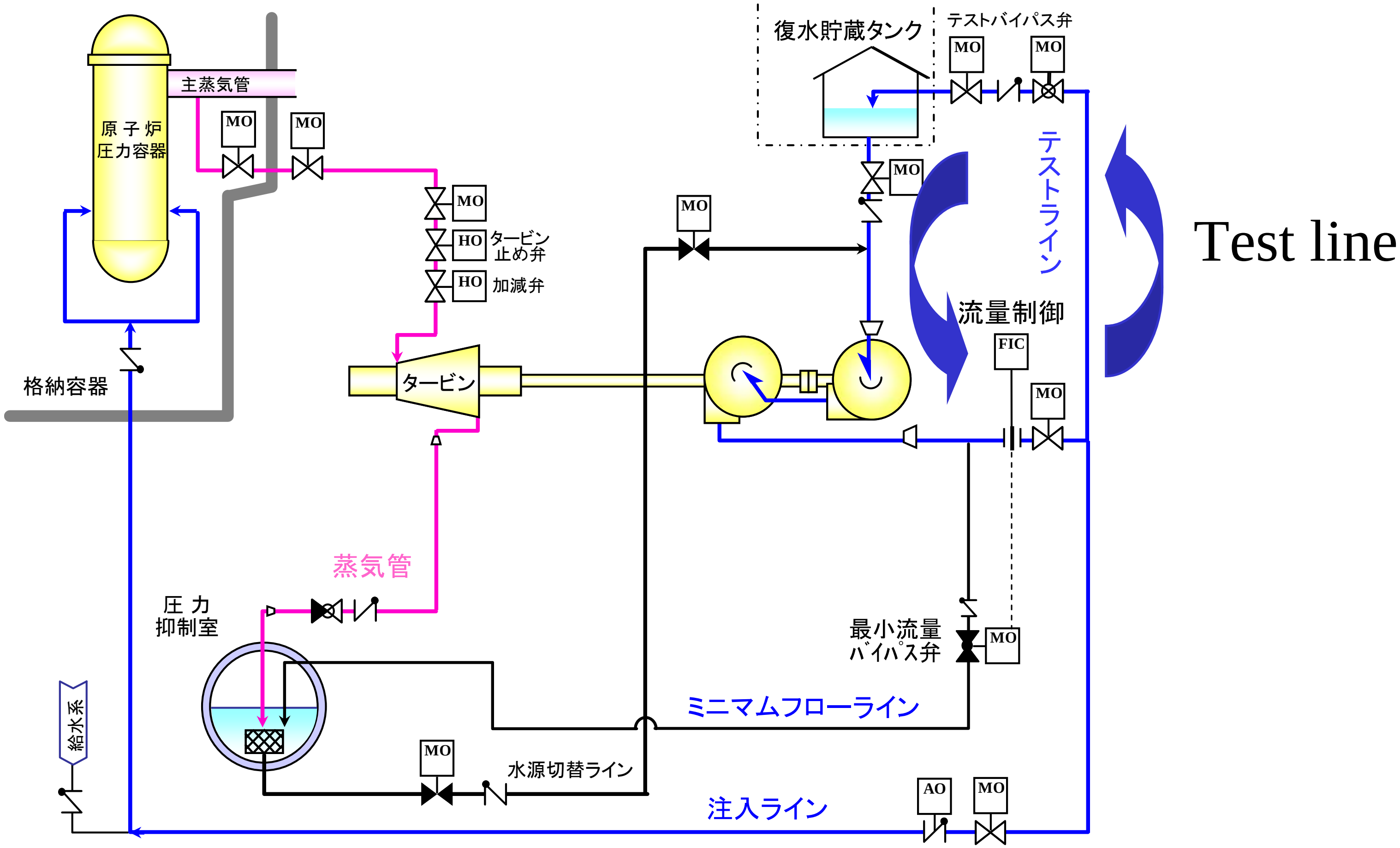
Calculated value of RPV pressure does NOT agree with measurement.

During pressure drop, step-like behavior is observed.

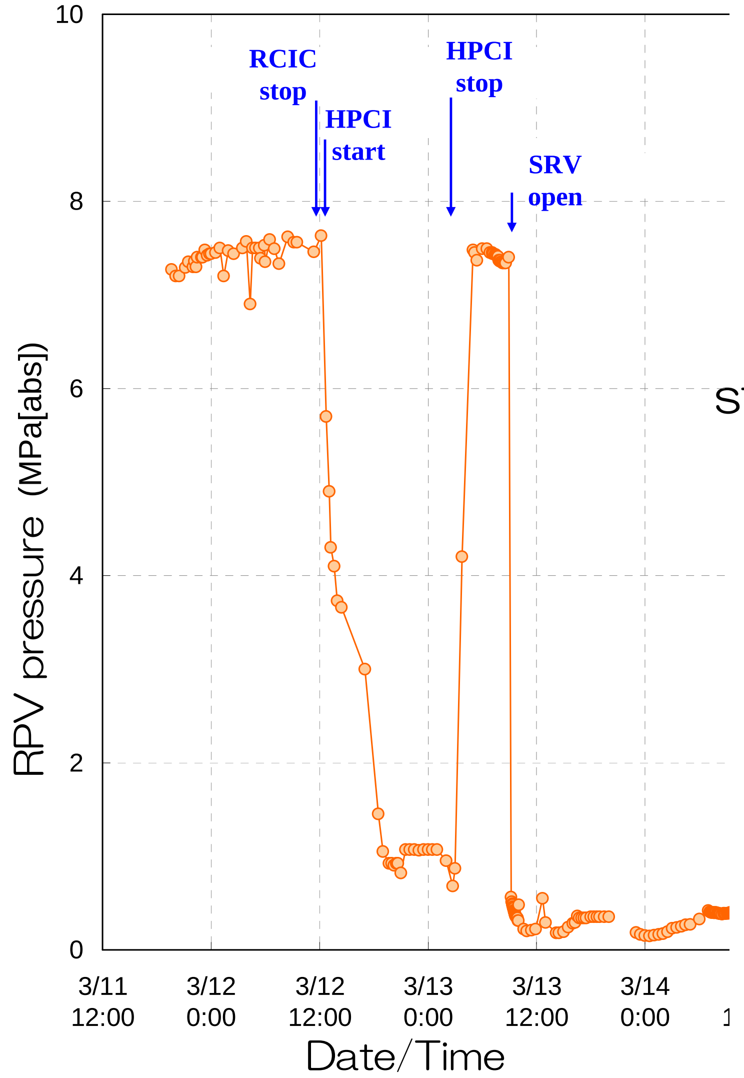


4．2．3号機のHPCIの運転操作

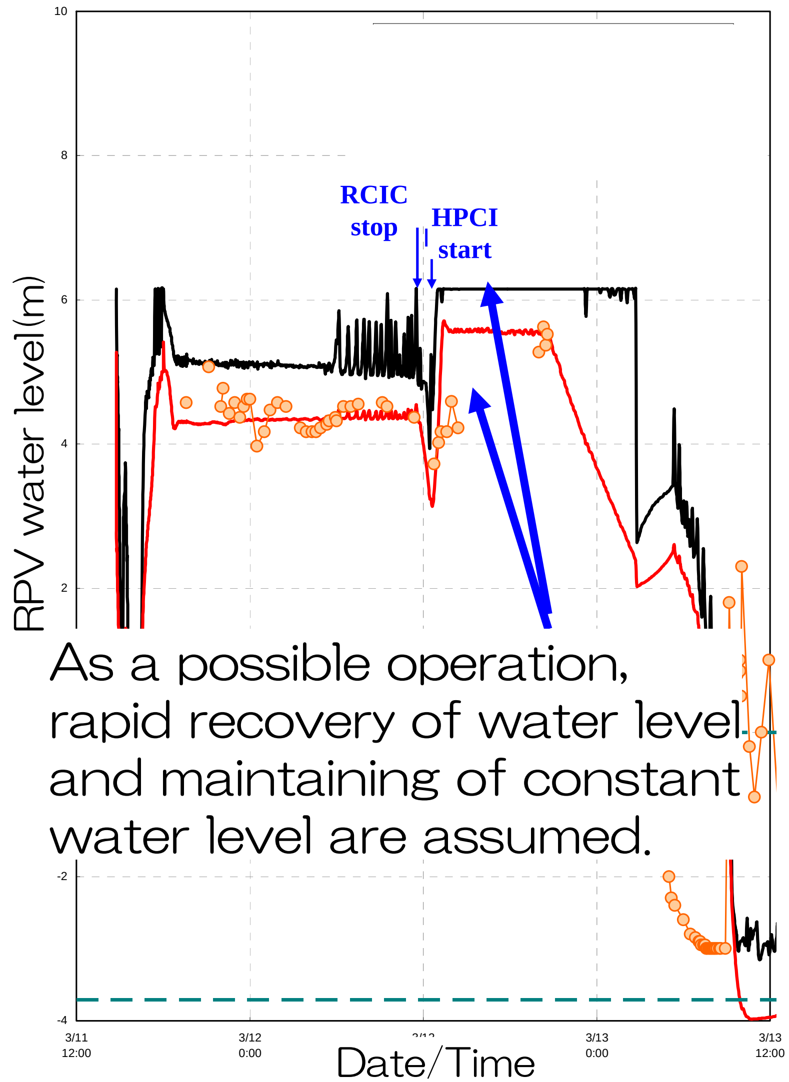
- From the interview with operators, It is clarified that HPCI had been continuously operated under control by using test line.
- Around the same time, S/C spray was conducted by diesel driven fire pump.



4. 2. HPCIの流量調整



Operation state change of HPCI?



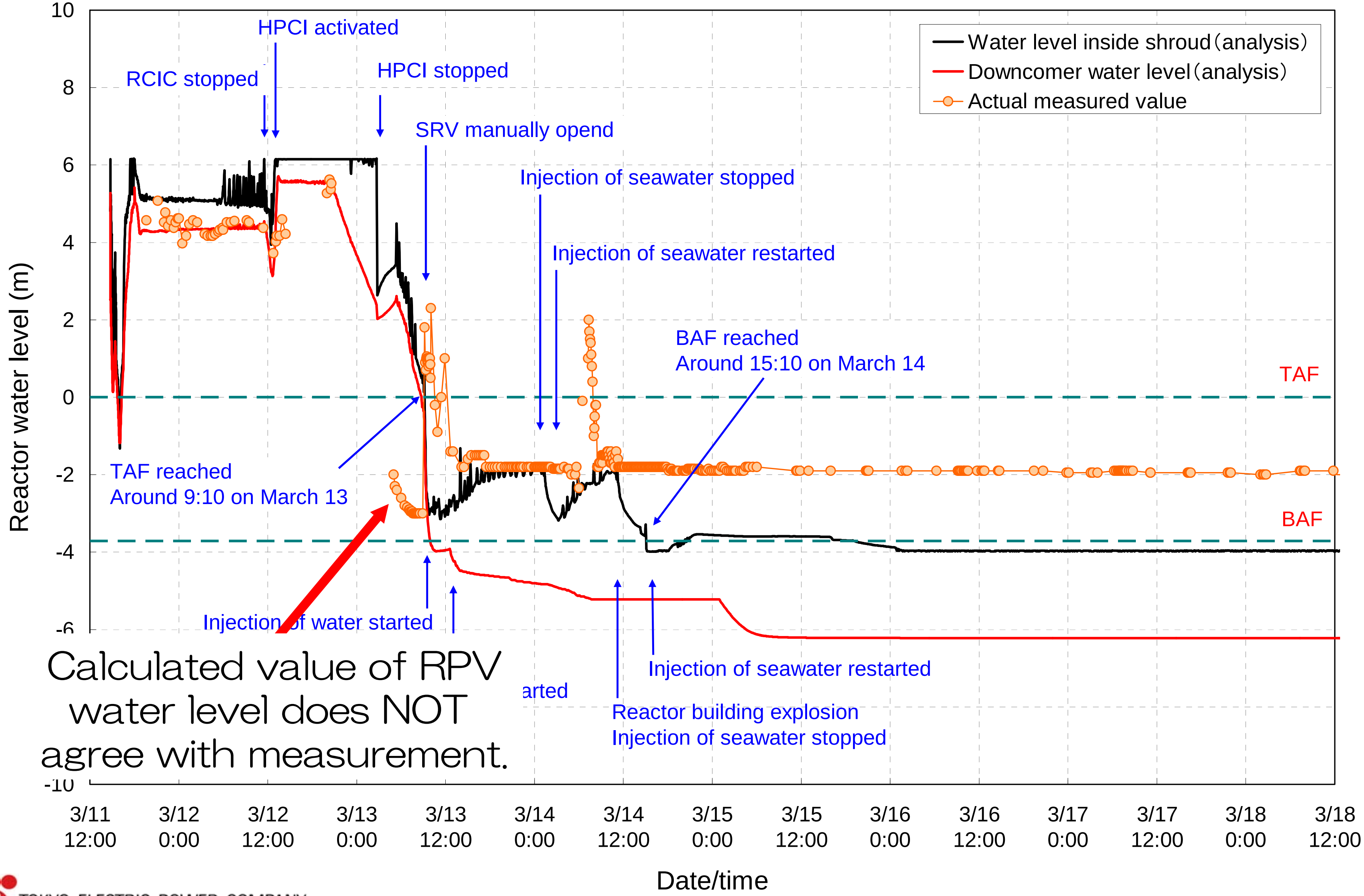
As a possible operation, rapid recovery of water level and maintaining of constant water level are assumed.

4．2．解析で採用した主な仮定（3号機）

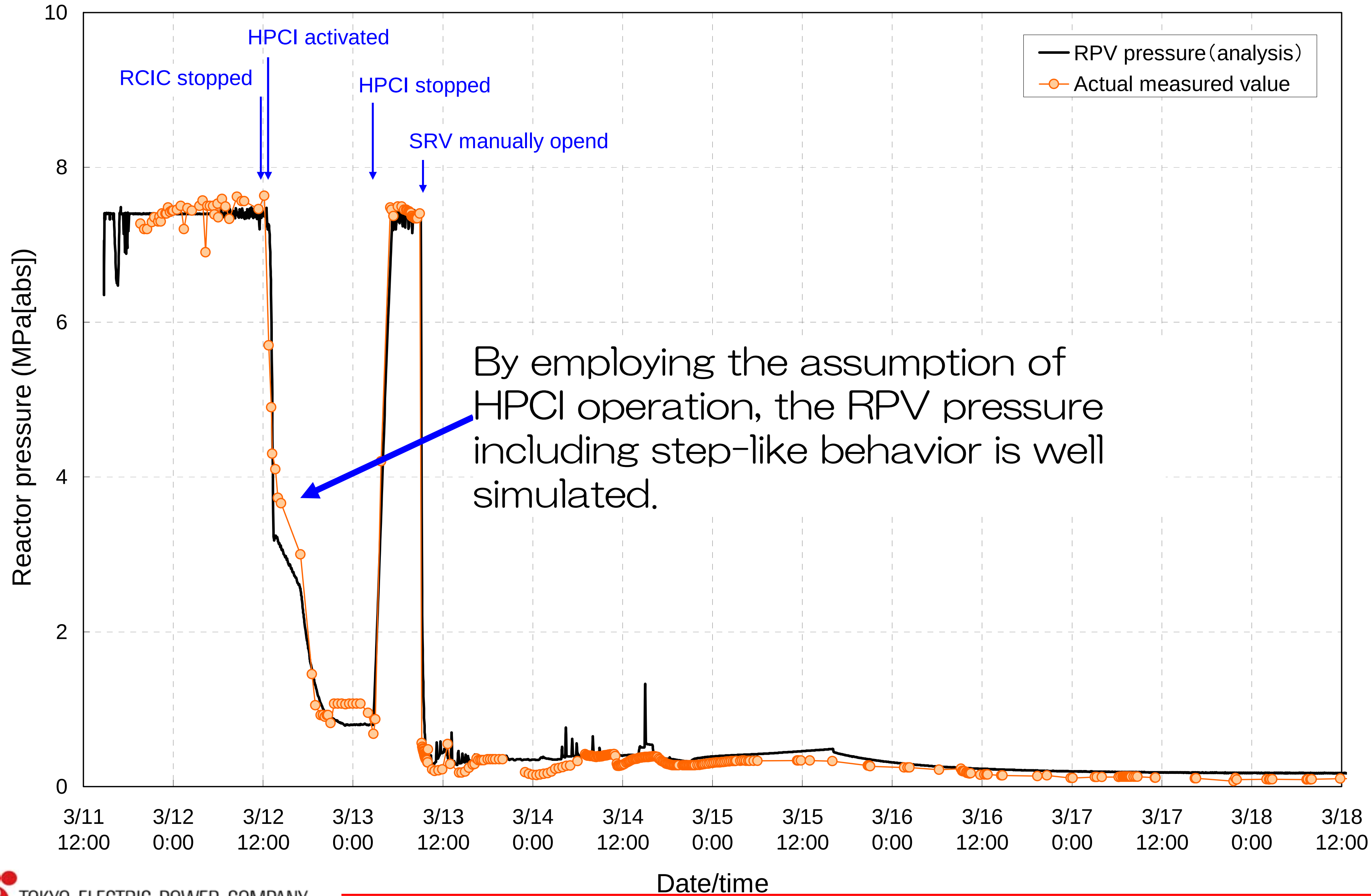
Assumptions

- RCIC and HPCI will be operated continuously under operator's control to avoid start-stop operation due to high or low trip water level. ← issue(6),(9)
- Realistic decay heat reflecting fuel loading history is employed. ← issue(7)
- S/C spray will be activated by diesel driven fire pump. (actual operation) ← issue(7)
- The amount of injection water from fire trucks is set to keep water level around bottom of active fuel

4．3. 解析結果（原子炉水位:3号機）



4．3. 解析結果（原子炉圧力:3号機）



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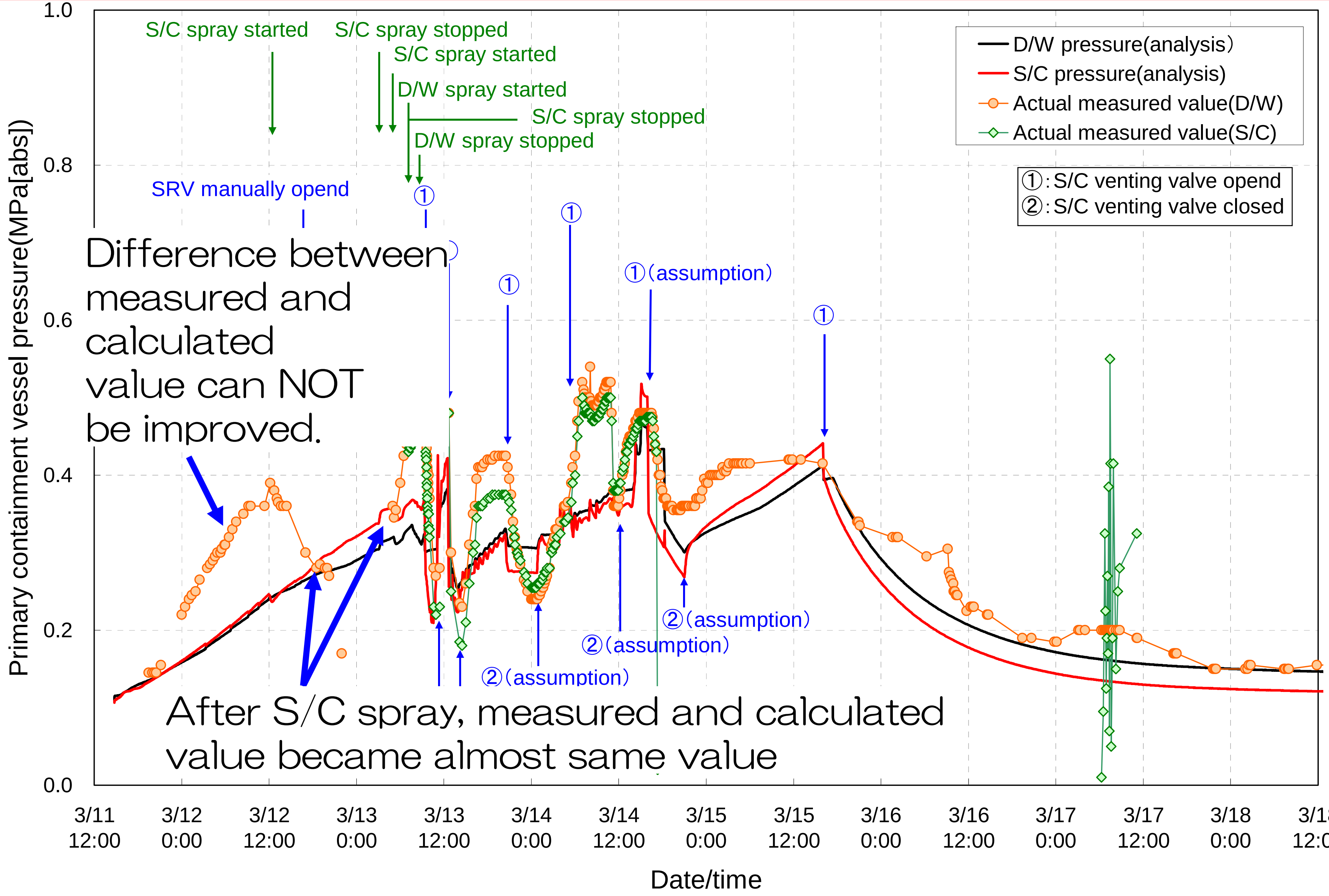
4．3. 解析結果（格納容器圧力:3号機）



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4．3．解析結果のまとめ（3号機）

<Summary>

・Water level reaches to TAF：3/13 09:10 （Approx. 42 hours after earthquake）

・Reactor core damage：3/13 10:40 （Approx. 44 hours after earthquake）

・RPV damage：NOT damaged

- As for issue 6 and 9: RPV pressure including step-like behavior is well simulated.
- As for issue 7: however difference between measured and calculated value of PCV pressure can NOT be improved, measured and calculated value of PCV pressure became almost same value after S/C spray.
- There are some possibilities that stratification in the S/C water caused excessive PCV pressure rise. (JNES had reported on 2012/02/01.)
- Reactor core was damaged, but RPV was NOT damaged. (So many information derived from operation conducted in Fukushima site until today indicates a high likelihood of RPV damage.)

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5. 今後の課題 1 : 炉心溶融後の評価

- BWR has complex structure in RPV bottom
  such as fuel support, CR guide tube.
- Every SA code has no detailed BWR bottom head model.

Modification of bottom head model is essential for improvement

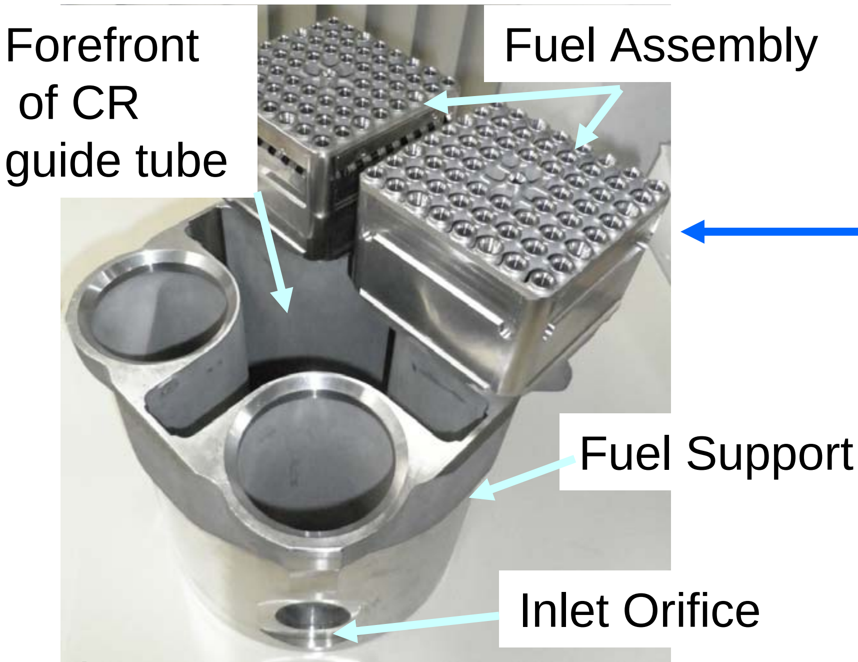


Fig. 1 BWR lower core structure

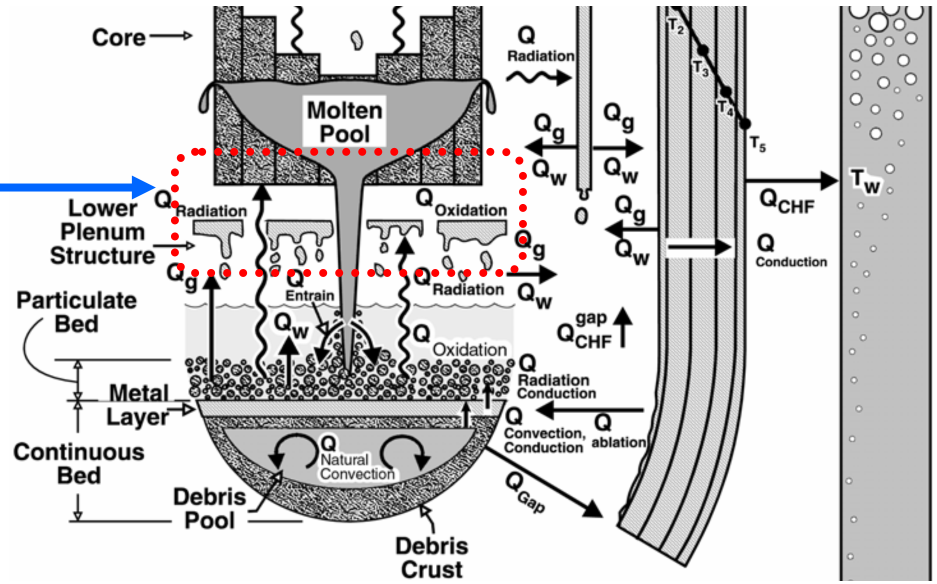
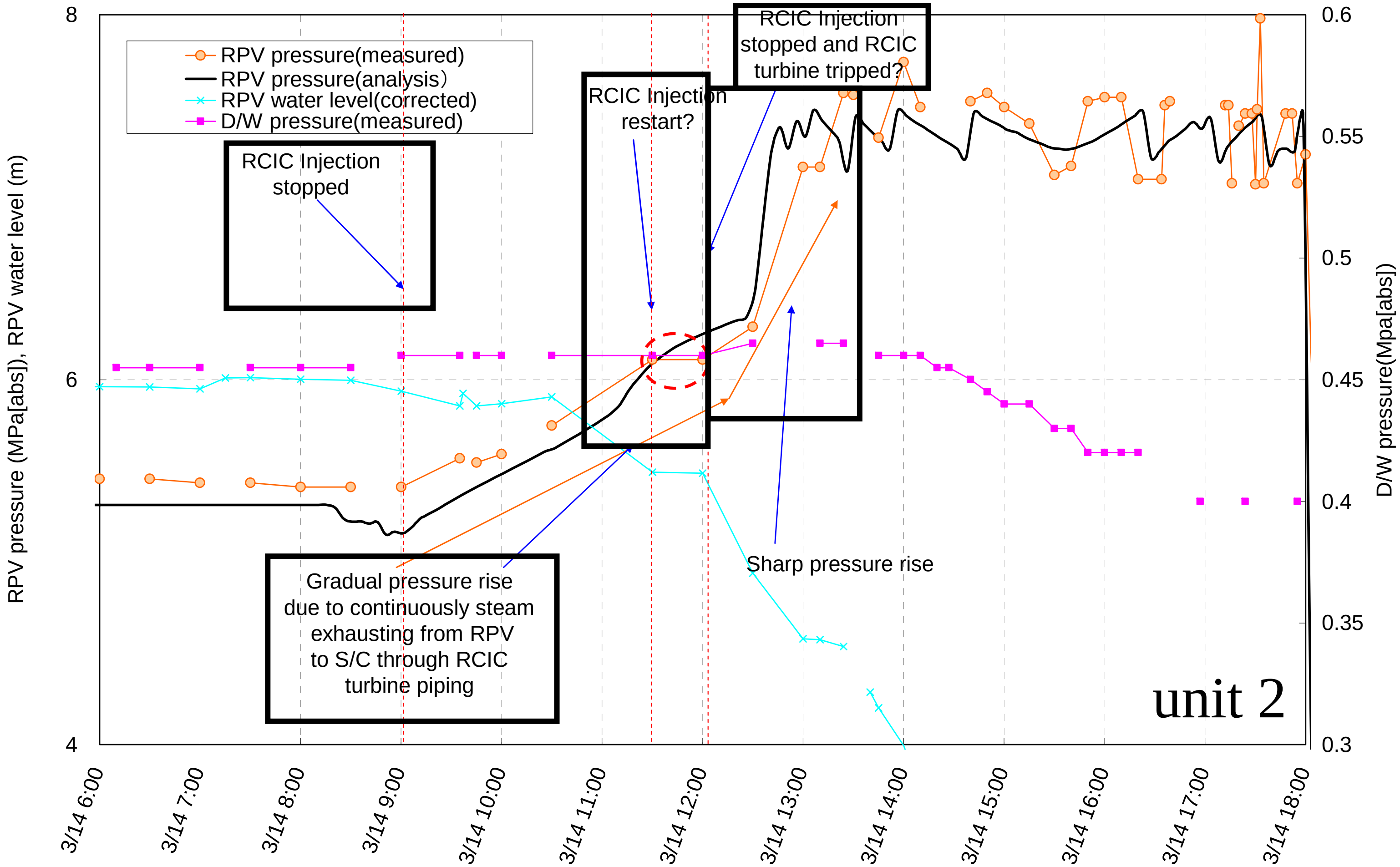


Fig. 2 MAAP model for BWR lower core structure

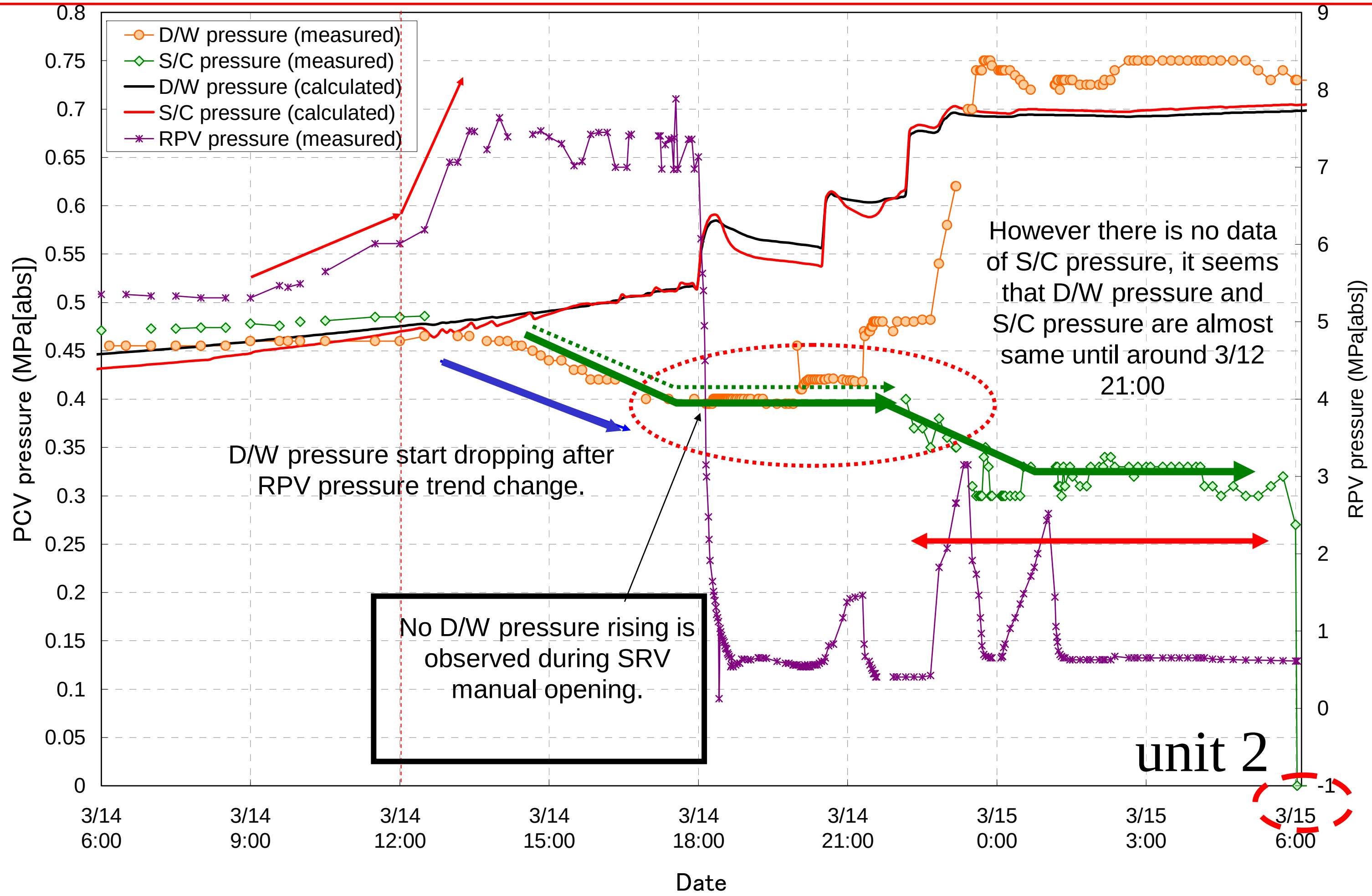
5.今後の課題２：2号機のD/W、S/C圧力挙動（1/2）



・Regarding to RCIC operation, we are now considering a following scenario;

- -RCIC injection stopped at about 3/14 9:00, but steam flowed into RCIC turbine continuously.
- -(Note: In the official report, RCIC stopping was confirmed by observation of water level decrease.)
- -For some reason, RCIC injection restarted around 3/14 11:30
- -Maybe due to over speed, RCIC turbine was mechanically tripped and RCIC injection was stopped.

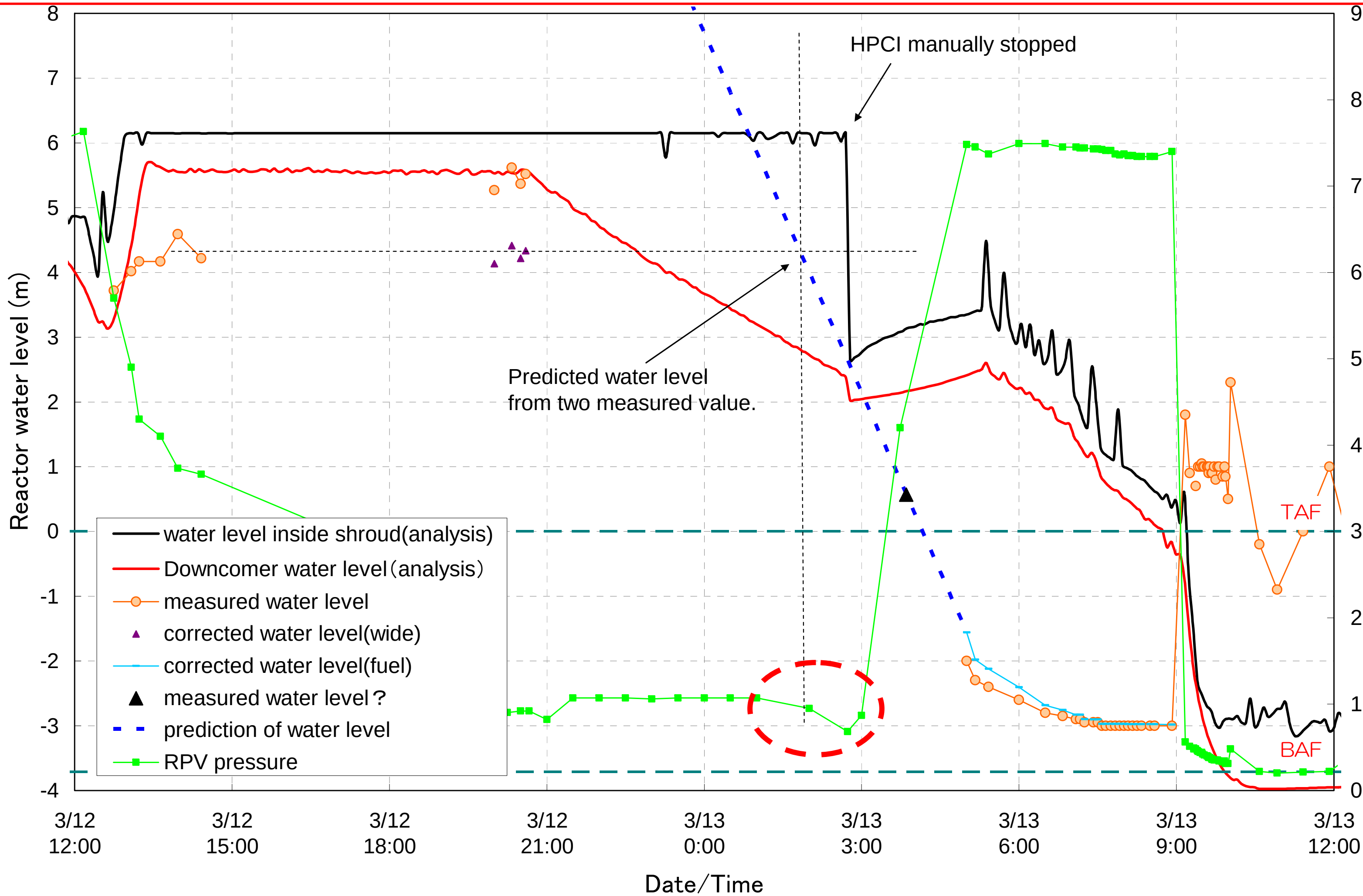
5.今後の課題２：2号機のD/W、S/C圧力挙動（2/2）



- It can be described the behavior of D/W pressure drop qualitatively. It is not possible to say that relatively low value of S/C pressure from 3/15 0:00 to 6:00 was due to S/C pressure gage malfunction.
- Because of indication of 0Pa[abs](downscale), It is obvious that S/C pressure gage was malfunctioned at 3/15 6:00.
- S/C temperature measured from 3/14 13:00 to 15:30 was under saturated temperature.

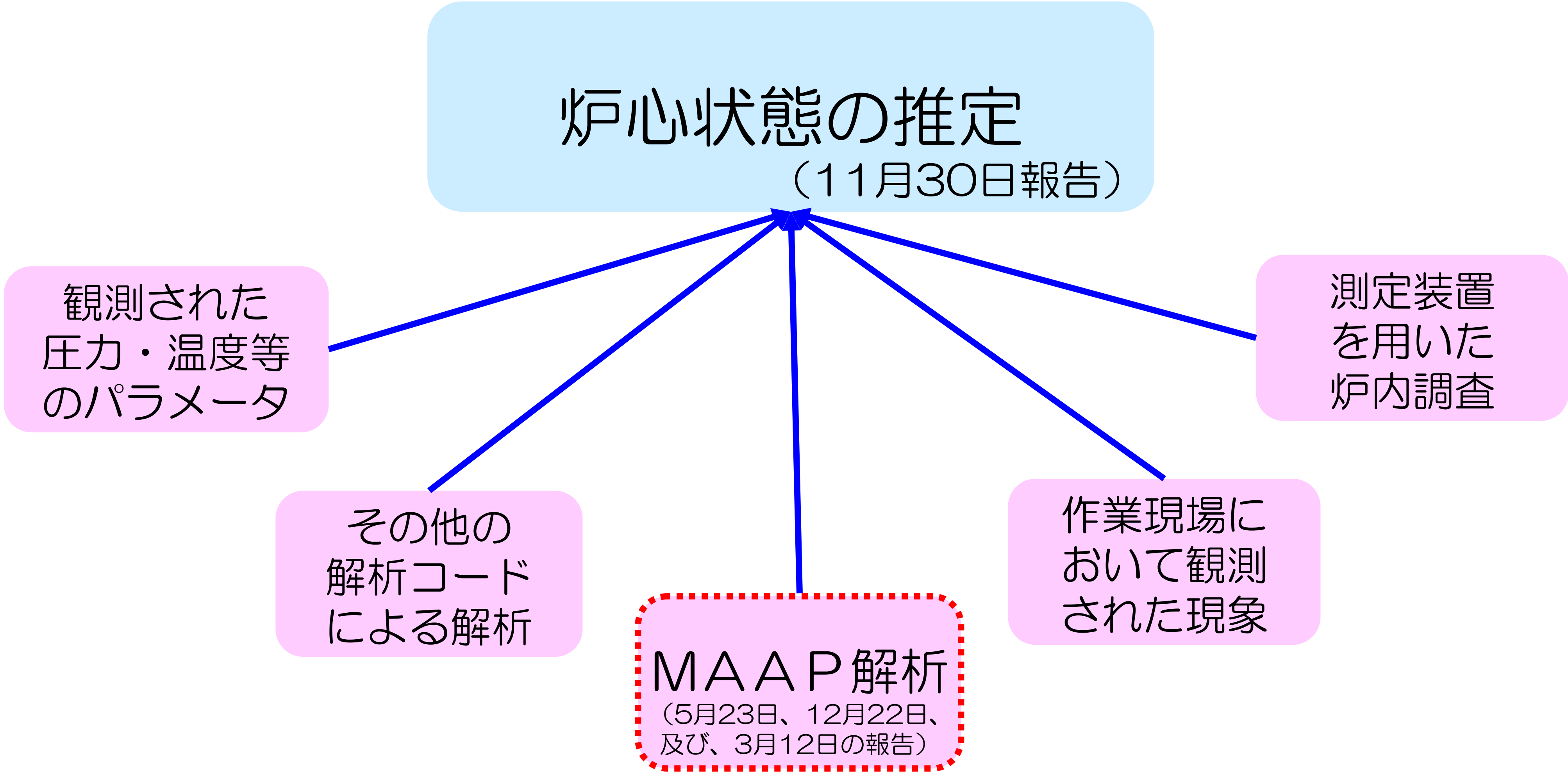
http://www.tepco.co.jp/en/nu/fukushima-np/plant-data/f1_2_Chart2.pdf P15,16

5.今後の課題３：3号機のHPCIの運転状態



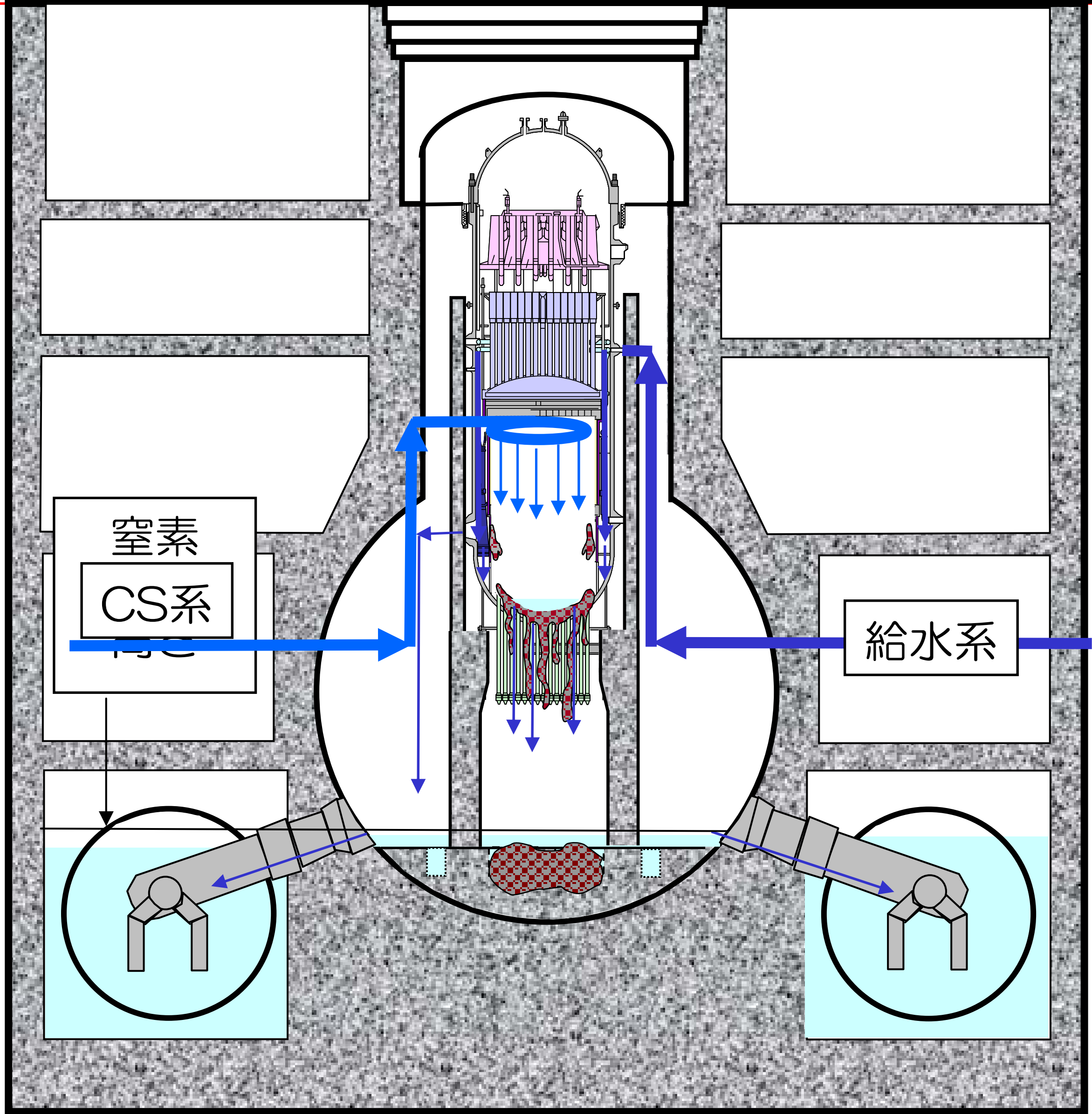
- In the time line, HPCI was stopped by operator at 2:42 3/13.
- Comparing measured value to calculated value, it seems that HPCI injection did NOT work before manual stop. In this period, RPV pressure is lower than that of turbine design.

5. 炉心状態の推定とMAAP解析の関係



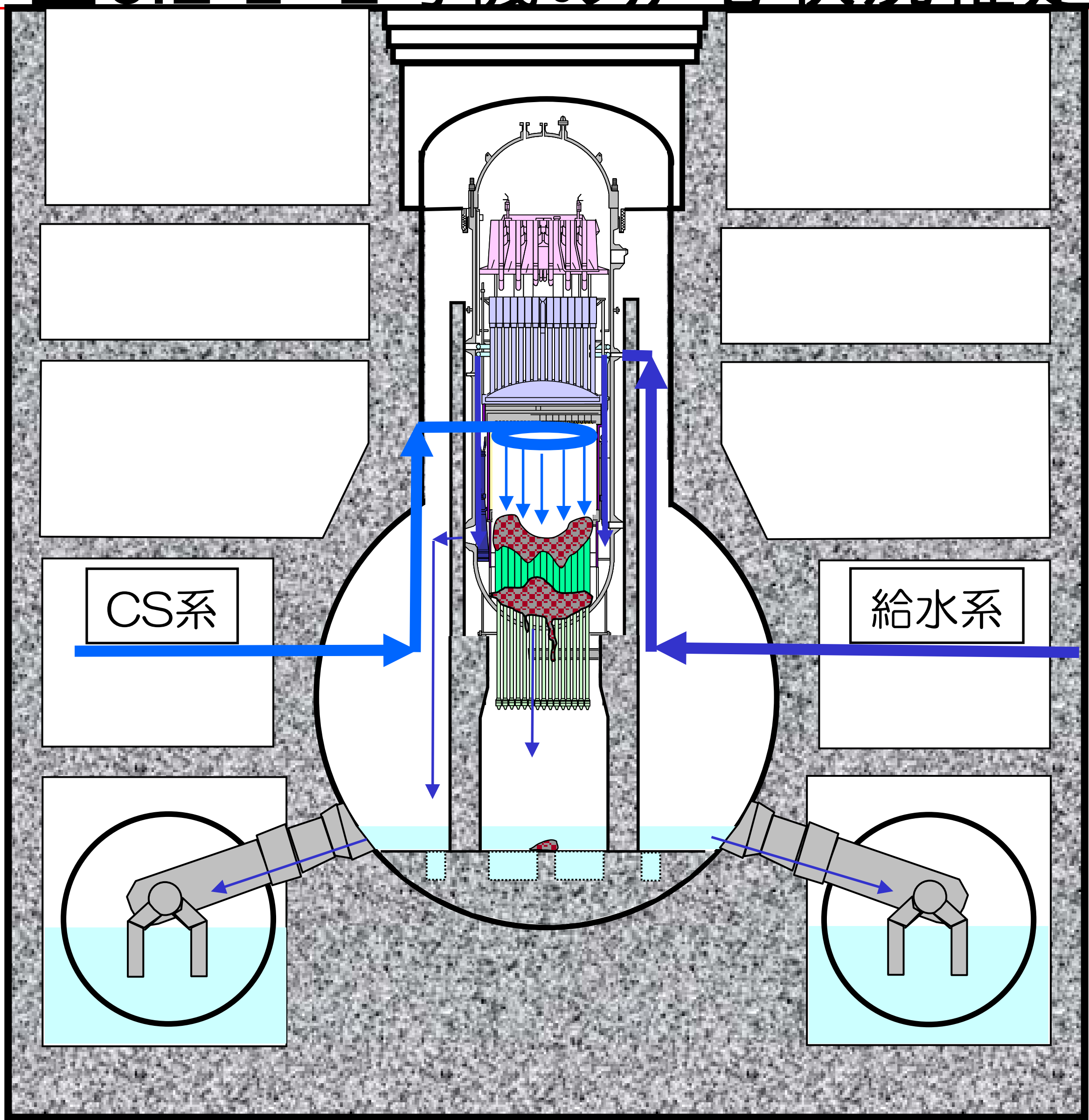
炉心状態の推定は、様々な情報を集約した上で、総合的に判断
（3/12のMAAP解析では、11/30報告の炉心状態の推定
の不確かさの幅を縮めるには至らなかった）

11月30日報告書
図6.1-1　1号機の炉心状況推定図



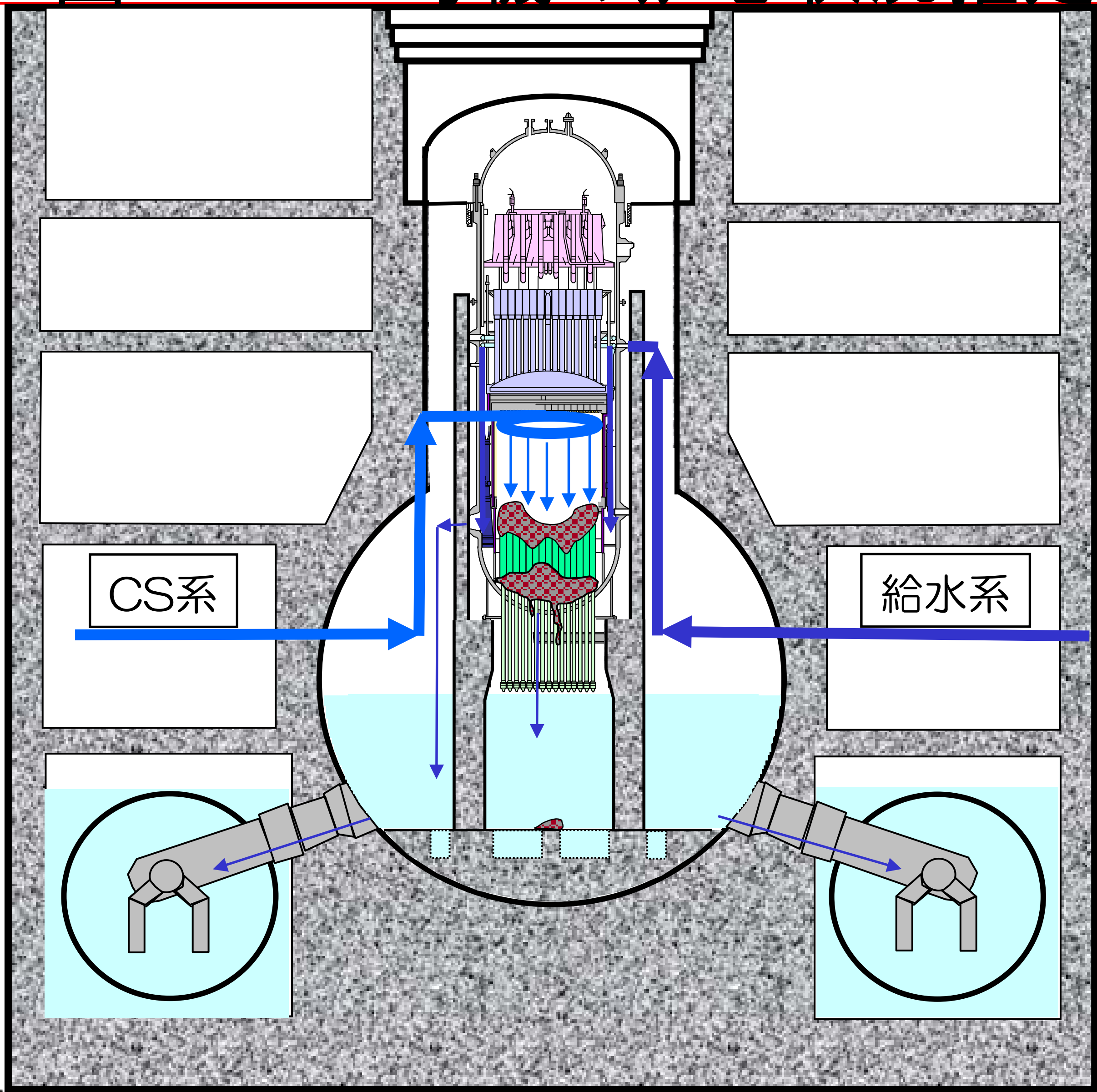
2011/12/10より、
CS系からの
注水も実施

11月30日報告書
図6.2-1 2号機の炉心状況推定図



内部調査により
水位約60cm
を確認
(2012/03/26)

11月30日報告書
図6.3-1　3号機の炉心状況推定図



2012/04/19、
機器ハッチ
（OP11340）
での水の滲み
を観測
(1階床OP10200)

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6. まとめ

- 現時点における、推定を含め明らかになっている情報（運転員による操作、プラントの特徴からの推定等）を元に、解析を実施した
- その結果、炉心溶融より前の段階については、概ね事故時のプラント挙動を再現することができた
- 実測値と整合しないいくつかの点について、今後の改善に向けての検討を実施した
- 他方で、本解析では２，３号機では圧力容器が破損しないという、観測されている事実と異なる結果となった。（炉心損傷状況の推定に関する技術ワークショップ（H23.11.30）報告参照）
- 現時点でのMAAPコードの解析能力の限界が明らかとなった。

- 政府・東京電力中長期対策会議は、研究開発推進本部の下に、炉内状況把握・解析サブワーキングチームを設置し、MAAPコードも含めたシビアアクシデント解析コードの高度化への取り組みを開始している

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7.（参考）使用済燃料プール事故評価の現状

- MAAP最新版（ver.5.0.1）、及び、MELCOR（ver.2.1）には、使用済燃料プールの水が失われる事故を評価出来るモジュールが存在
- 福島第一発電所においては、使用済燃料プールは冠水状態を維持することができたものの、リスク要因としては重要な課題
- モジュールの能力評価のため、試解析を実施
  - 解析作業は電力中央研究所が実施

7. （参考）解析条件(1/2)

- 各コードのSFP解析モデルを使用
- ORIGENにて評価した崩壊熱を再現するように崩壊熱計算式のパラメータを調整
- MELCOR解析は、取出し燃料をCh.1、その他使用済燃料をCh.2として設定
- MAAP解析は、出来る限りプール内の燃料配置を再現するよう、プール内の燃料ラックをグループ化して分割し、20Ch.を設定
- SFPから水が失われる解析シナリオとして、MELCOR解析では、1時間でプール水が喪失、MAAP解析では、水位0から解析を開始
- 蒸発により徐々に水が失われる場合も評価

7. （参考）解析条件(2/2)

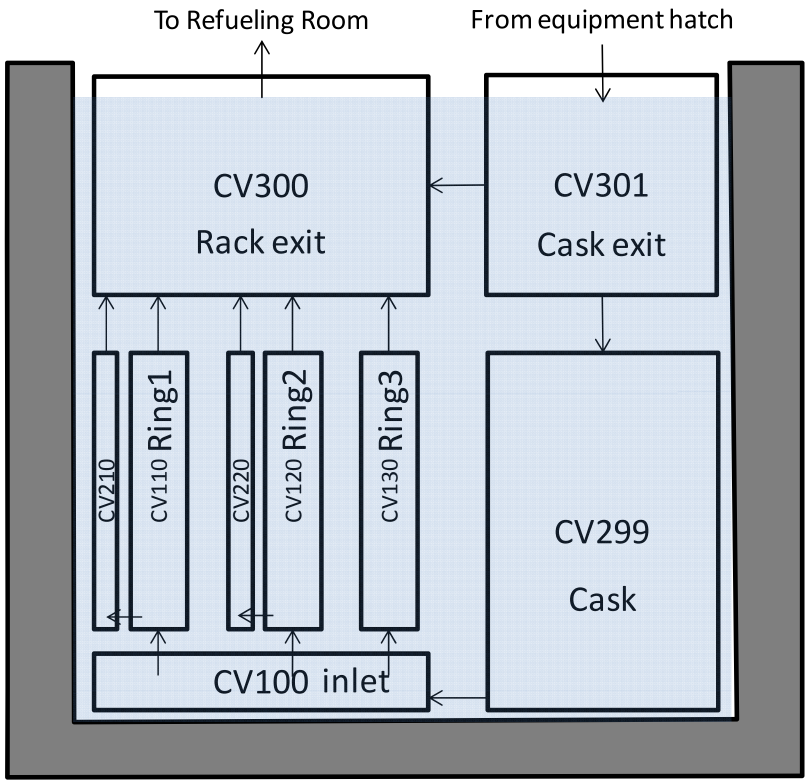


Fig.1 MELCOR model

取出し燃料と使用済み燃料でチャンネルを分割

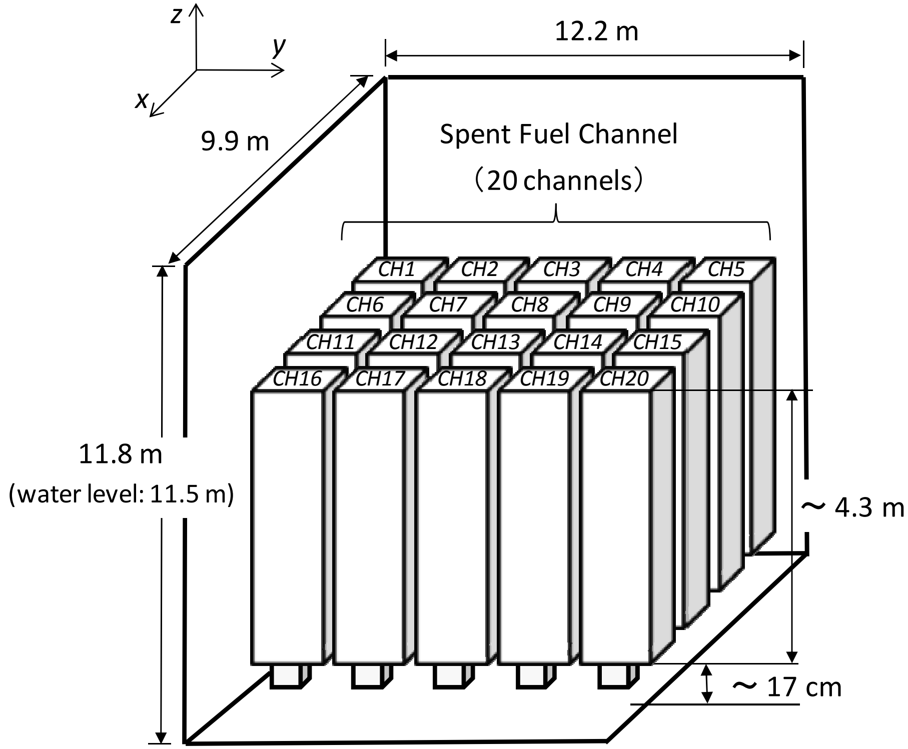
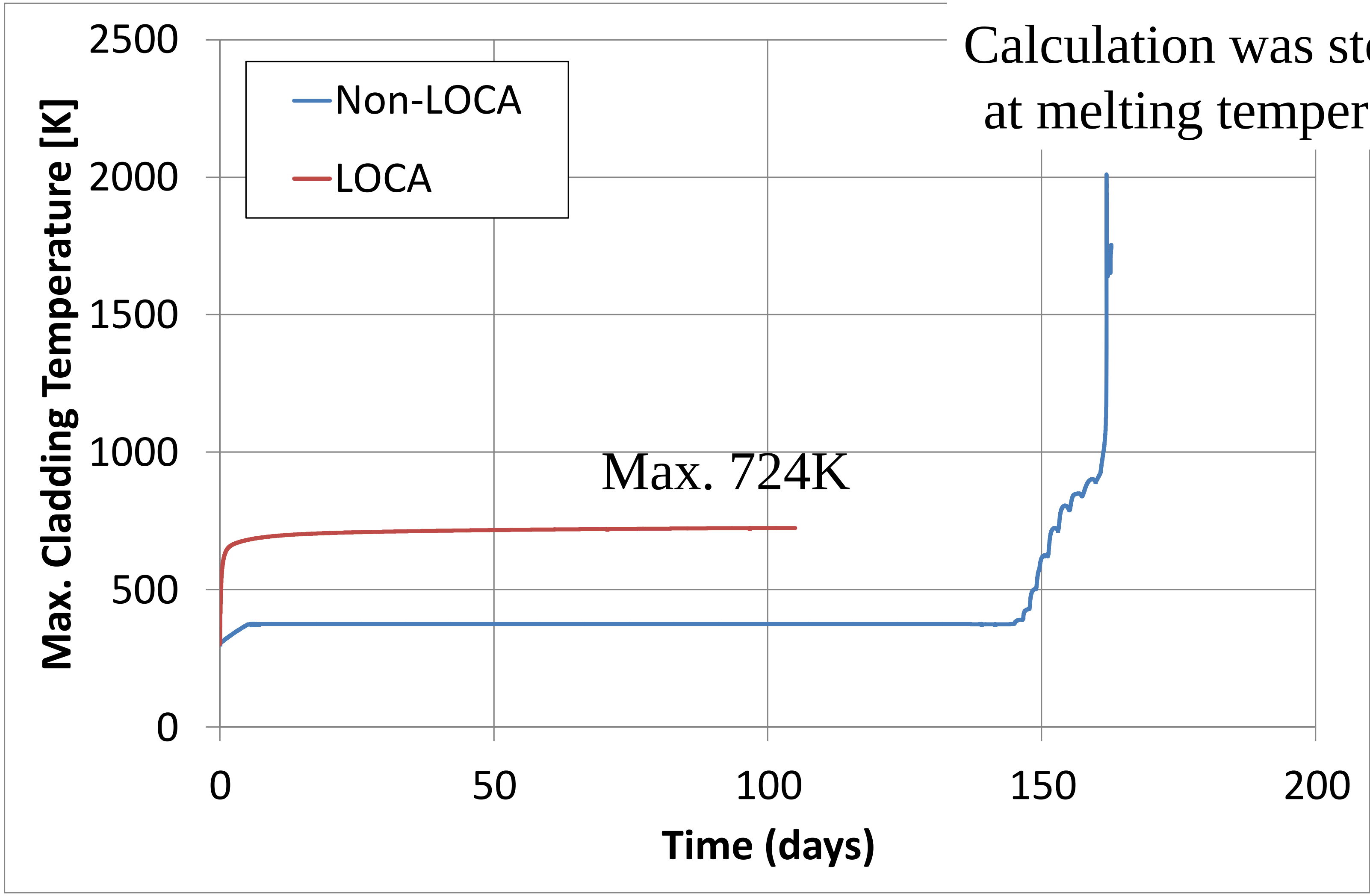


Fig.2 MAAP model

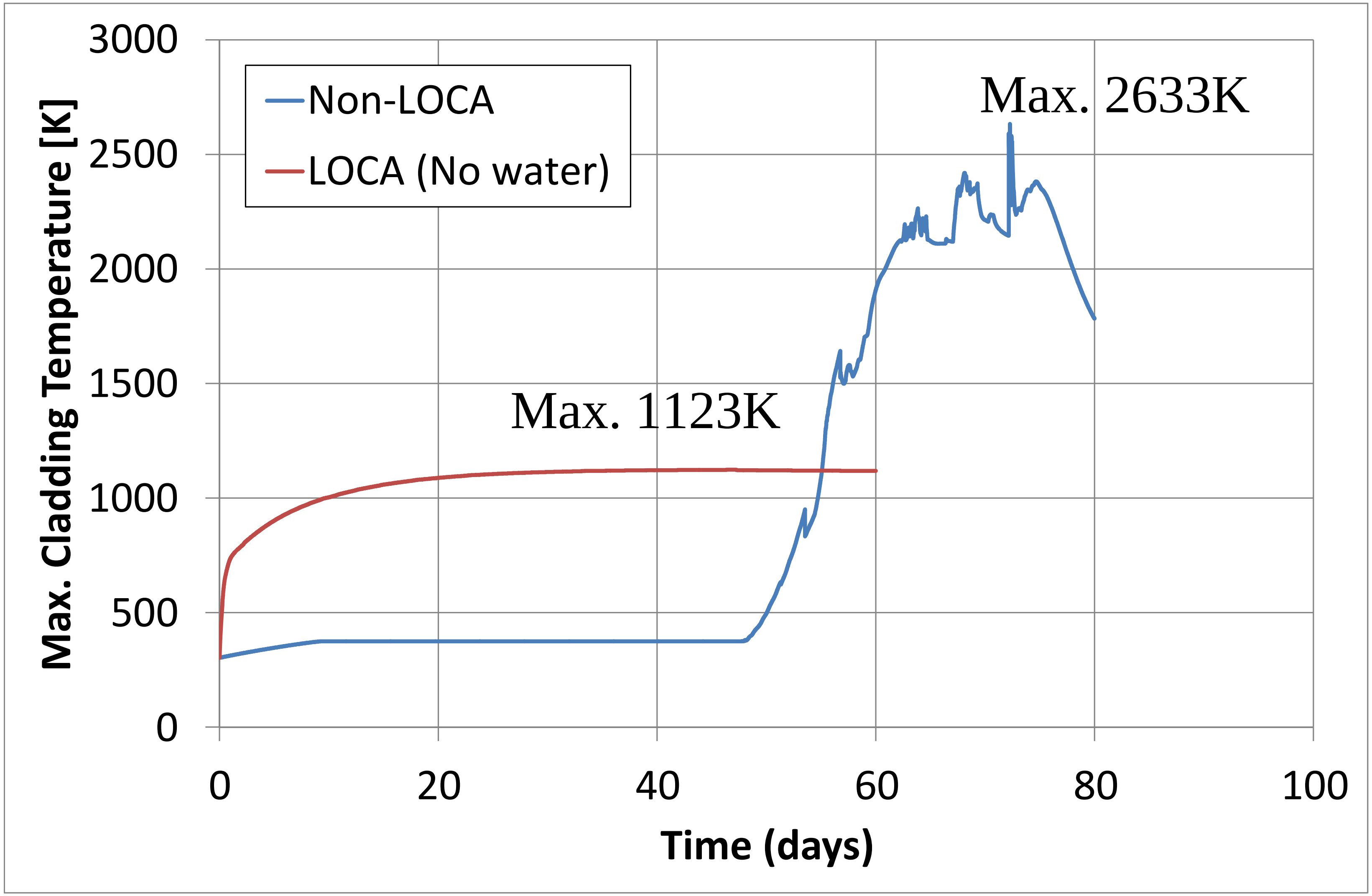
実際の燃料配置を反映してチャンネルを分割

両者とも、炉心評価モデルを利用する形でモデル化（空気流入有り）

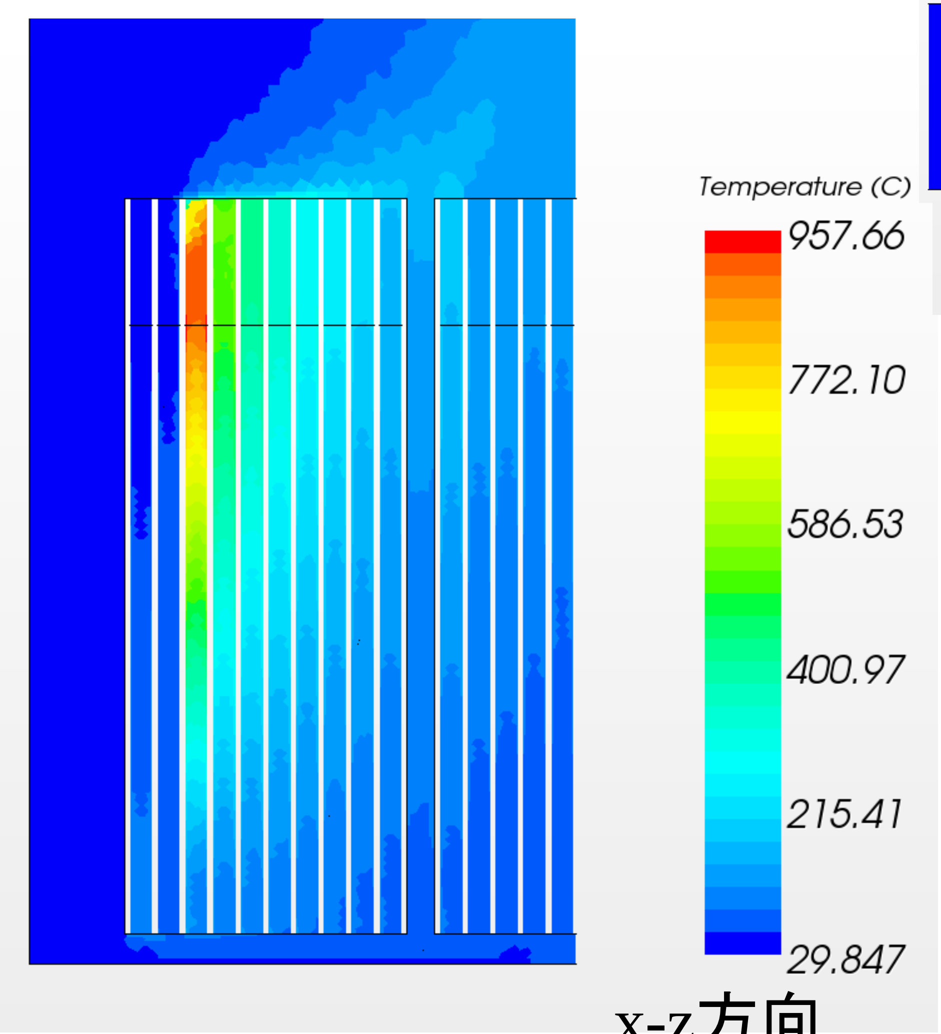
7. （参考）解析結果（MELCOR）



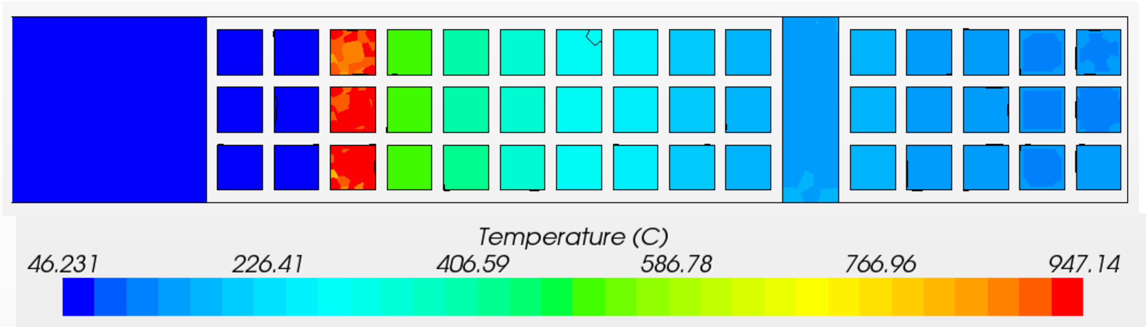
7. （参考）解析結果（MAAP）



7. （参考）空冷の場合のCFD評価



x-z方向
温度分布



x-y方向温度分布

CFD評価により、空冷のみの条件であっても、最大960℃程度の温度となると評価
（保守的に1500W/体にて評価）
水がない場合でも、空気冷却で崩壊熱分の除熱がなされるとの、MELCOR解析、MAAP解析とほぼ同等の結果となっている。

CFD評価はテプコシステムズが実施（STAR-CCM+使用）

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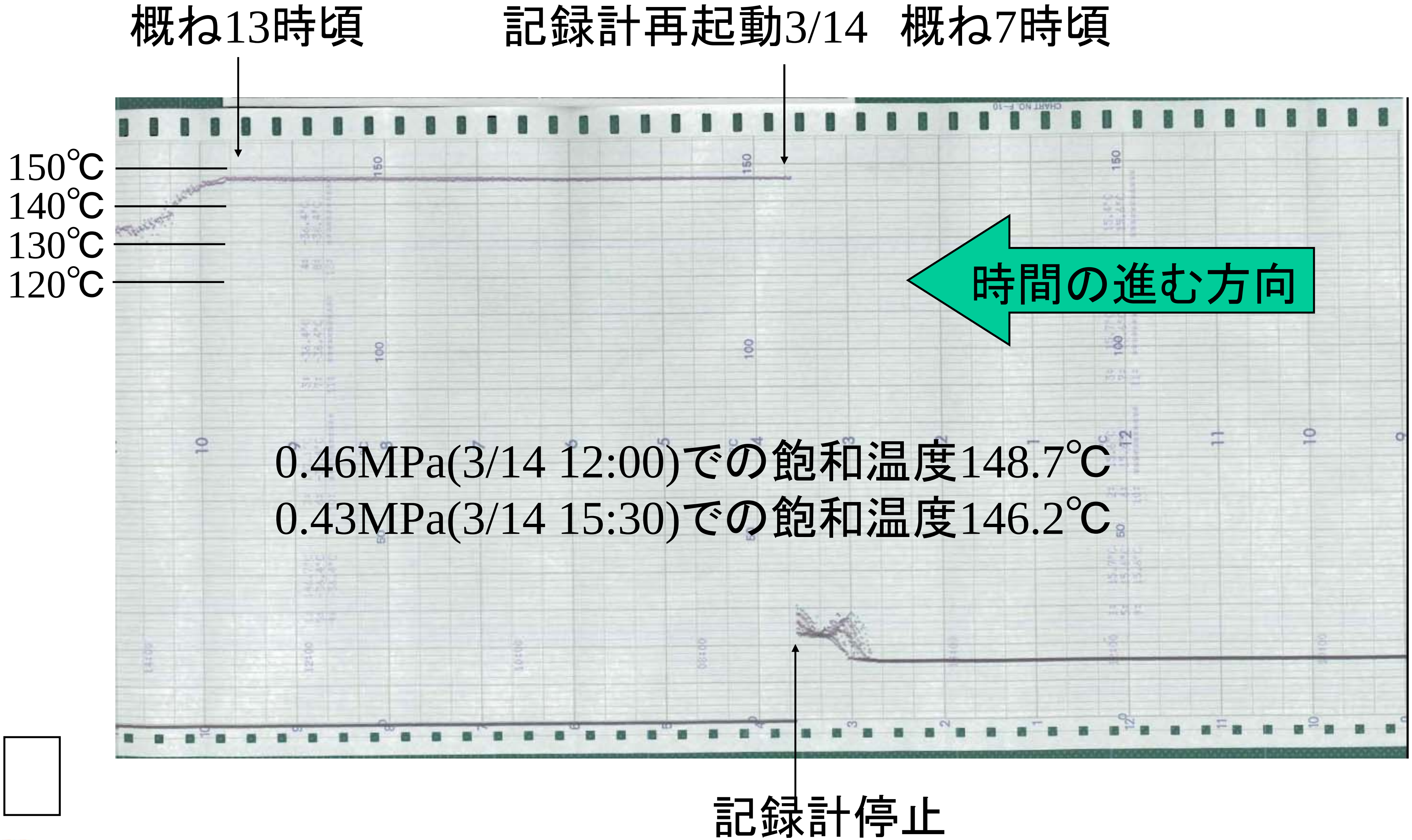
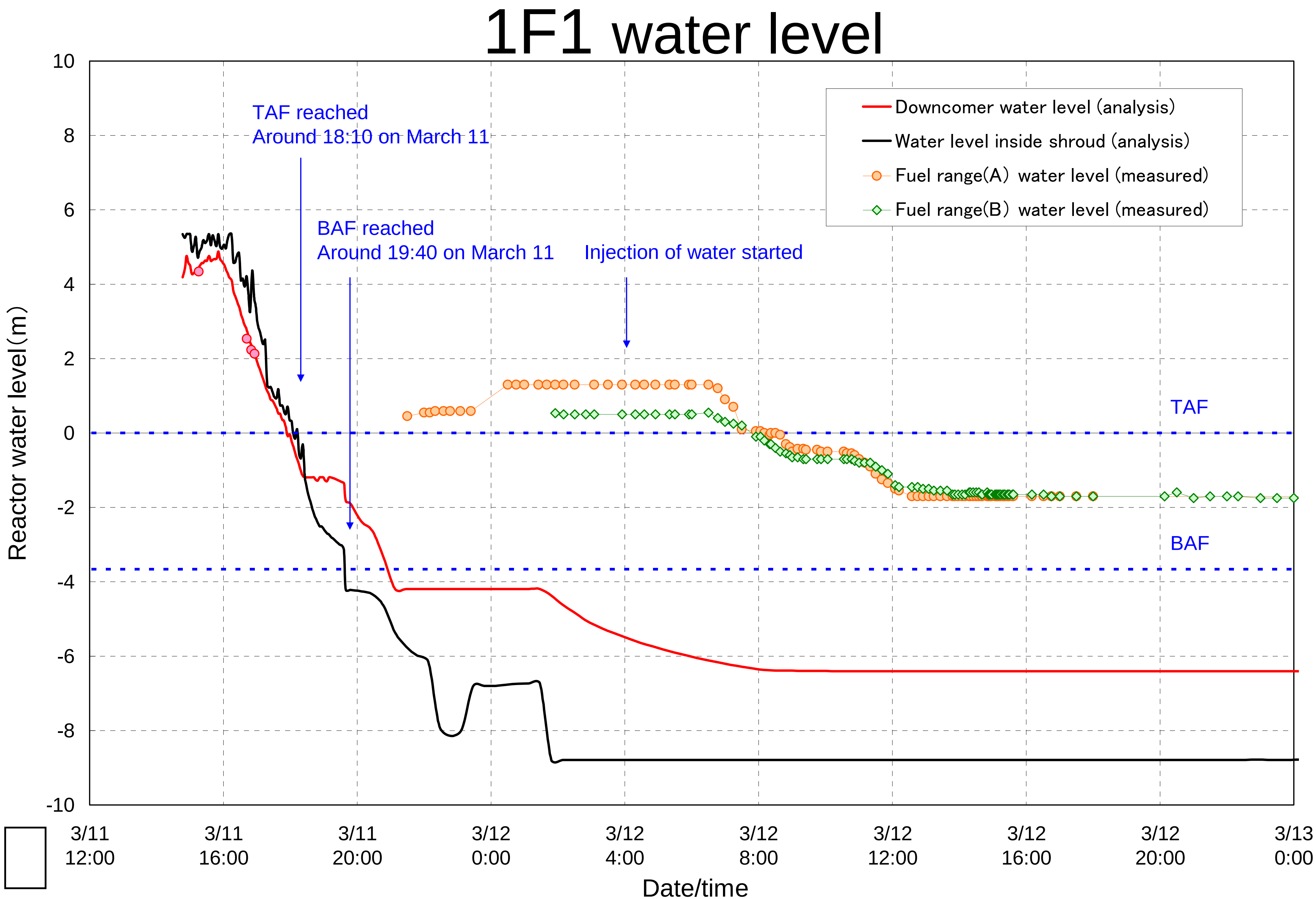
7. （参考）使用済燃料プール解析のまとめ

- 今後更なる検討が必要であるものの、今回の試評価では、どちらの解析コードを用いても、使用済燃料プールの水が急速に失われた場合には、空気冷却により約450～850℃程度で安定する結果
- プールの冷却・注水機能が喪失した場合に相当する蒸発による水喪失解析では、ジルコニウムの酸化反応により、温度が急上昇する結果

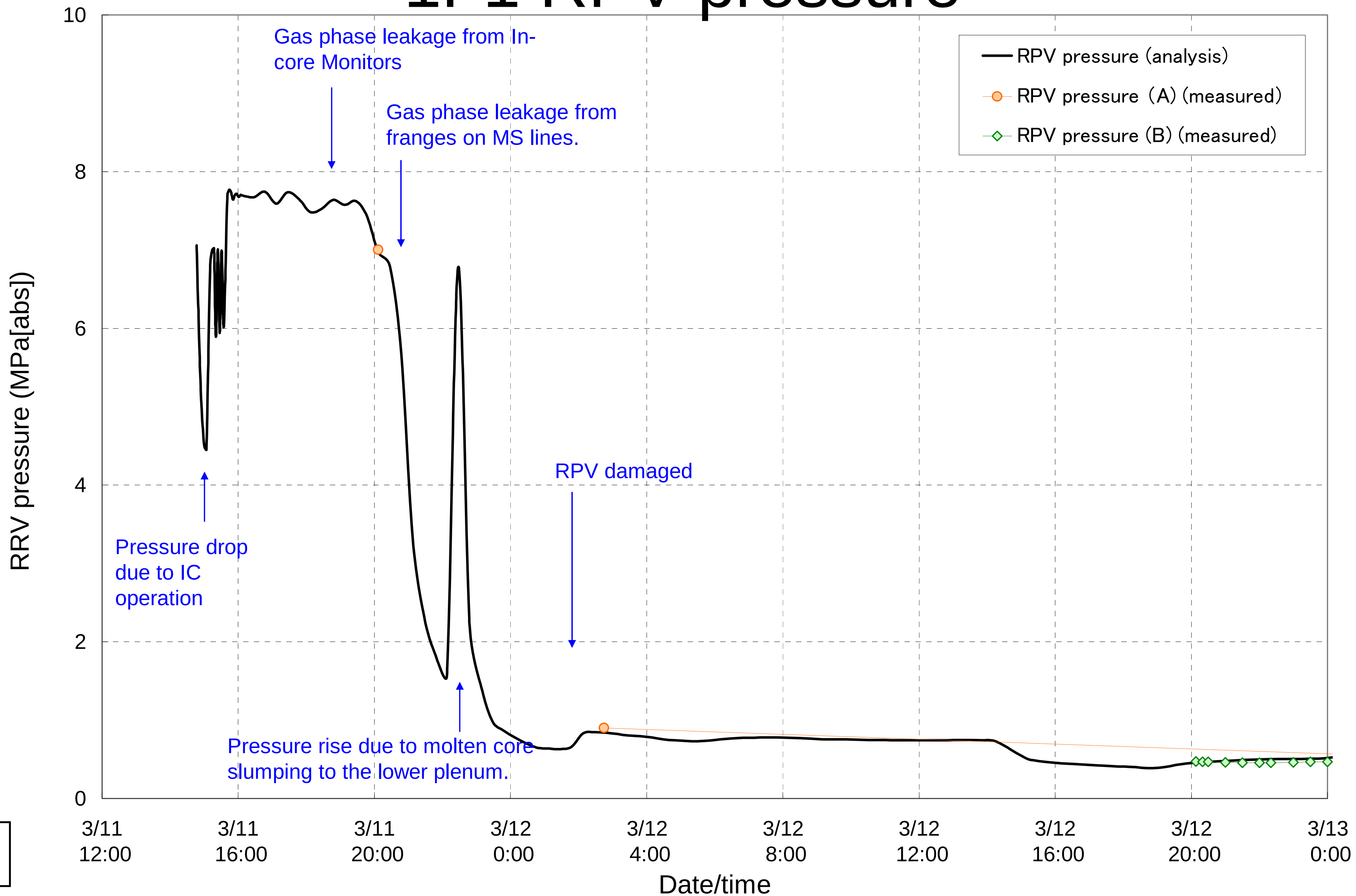
Fin.

チャート（S/C 温度）

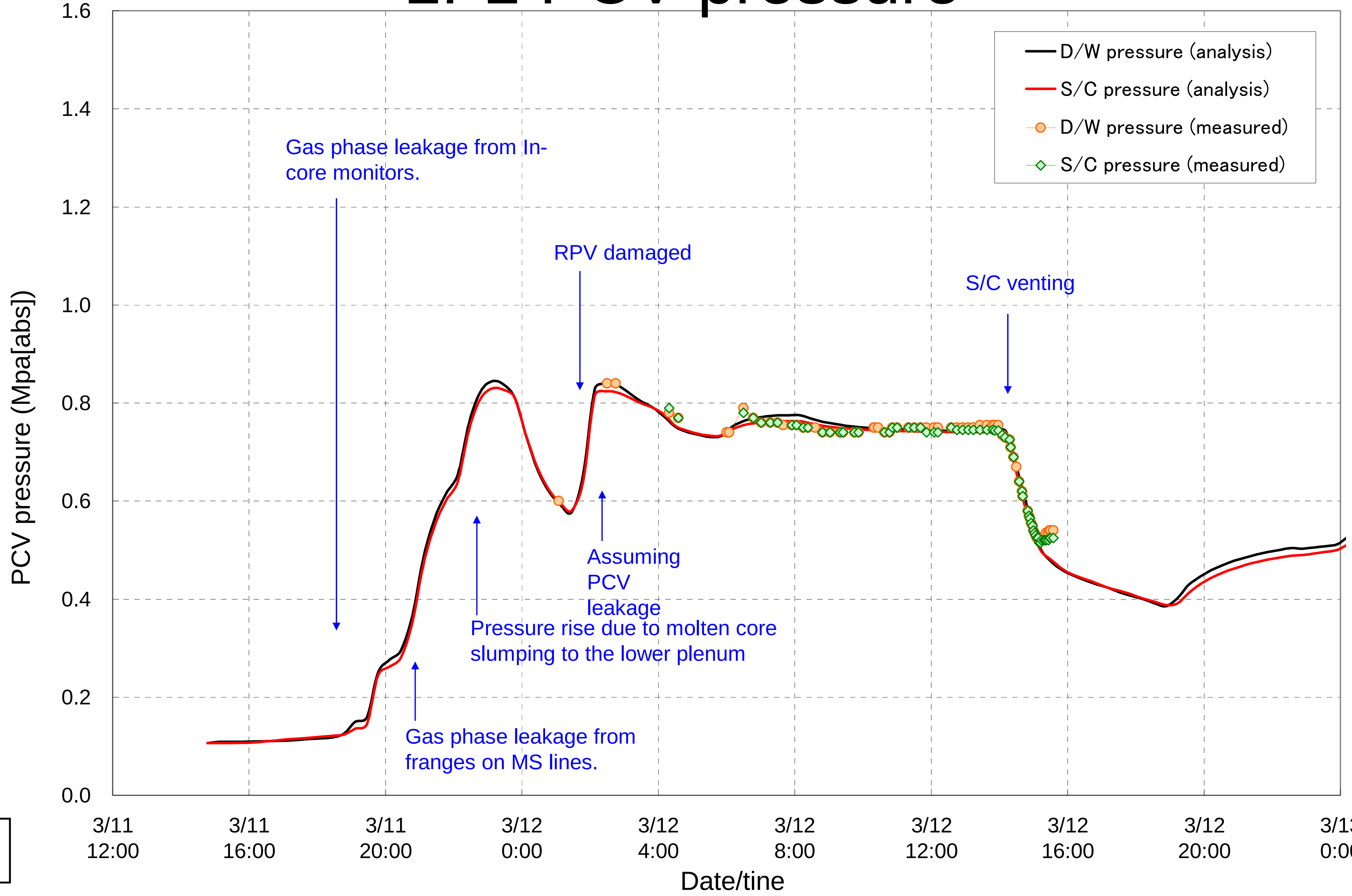

概ね13時頃
記録計再起動3/14　概ね7時頃
150℃
140℃
130℃
120℃
時間の進む方向
0.46MPa(3/14 12:00)での飽和温度148.7℃
0.43MPa(3/14 15:30)での飽和温度146.2℃
記録計停止


1F1 water level
Reactor water level(m)
10
8
6
4
2
0
-2
-4
-6
-8
-10
3/11 12:00
3/11 16:00
3/11 20:00
3/12 0:00
3/12 4:00
3/12 8:00
3/12 12:00
3/12 16:00
3/12 20:00
3/13 0:00
Date/time
TAF reached
Around 18:10 on March 11
BAF reached
Around 19:40 on March 11
Injection of water started
TAF
BAF
Downcomer water level (analysis)
Water level inside shroud (analysis)
Fuel range(A) water level (measured)
Fuel range(B) water level (measured)

1F1 RPV pressure



1F1 PCV pressure

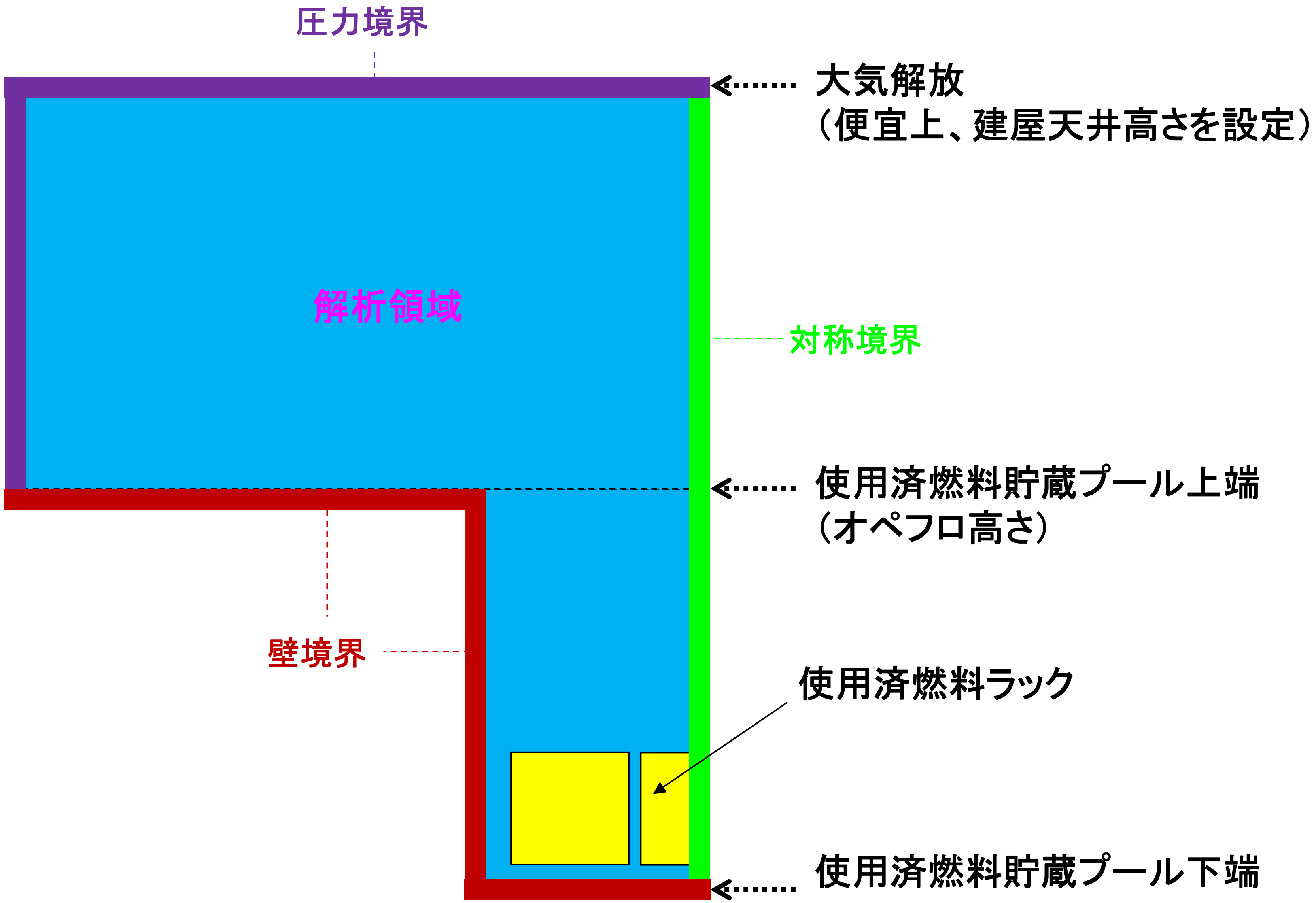


CFD評価モデル（１／２）

表1 評価条件

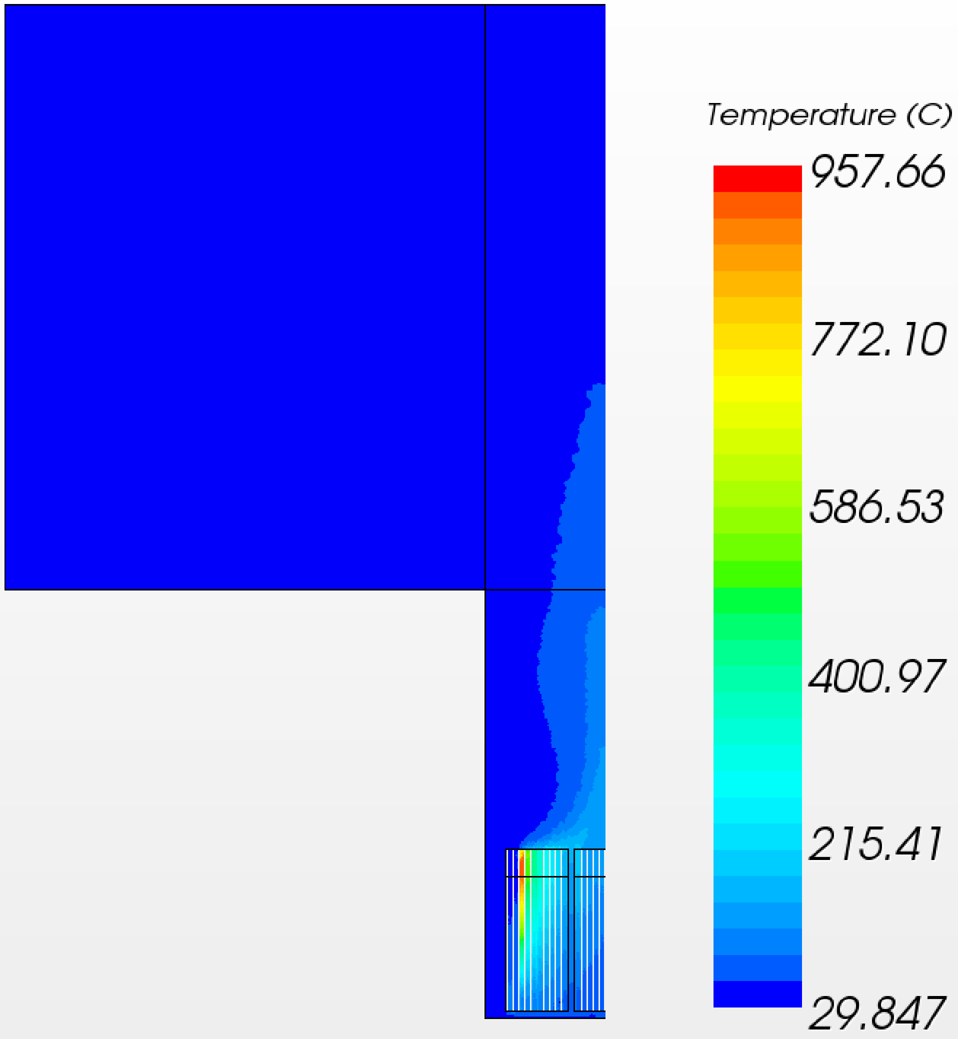
項目	設定	備考
時間領域	定常解析	
燃料集合体の崩壊熱	1500W (1体当たり)	2012/6/1時点において最も崩壊熱が高い燃料集合体を想定し、ラック内の燃料全てに左記の崩壊熱を設定。
境界温度	30℃	外気温を想定。
境界圧力	大気圧	
乱流モデル	k- ε モデル	Realizable k- ε 2層モデルを設定。
気体密度	温度依存	理想気体の温度依存性を考慮。
対流項評価精度	2次精度	

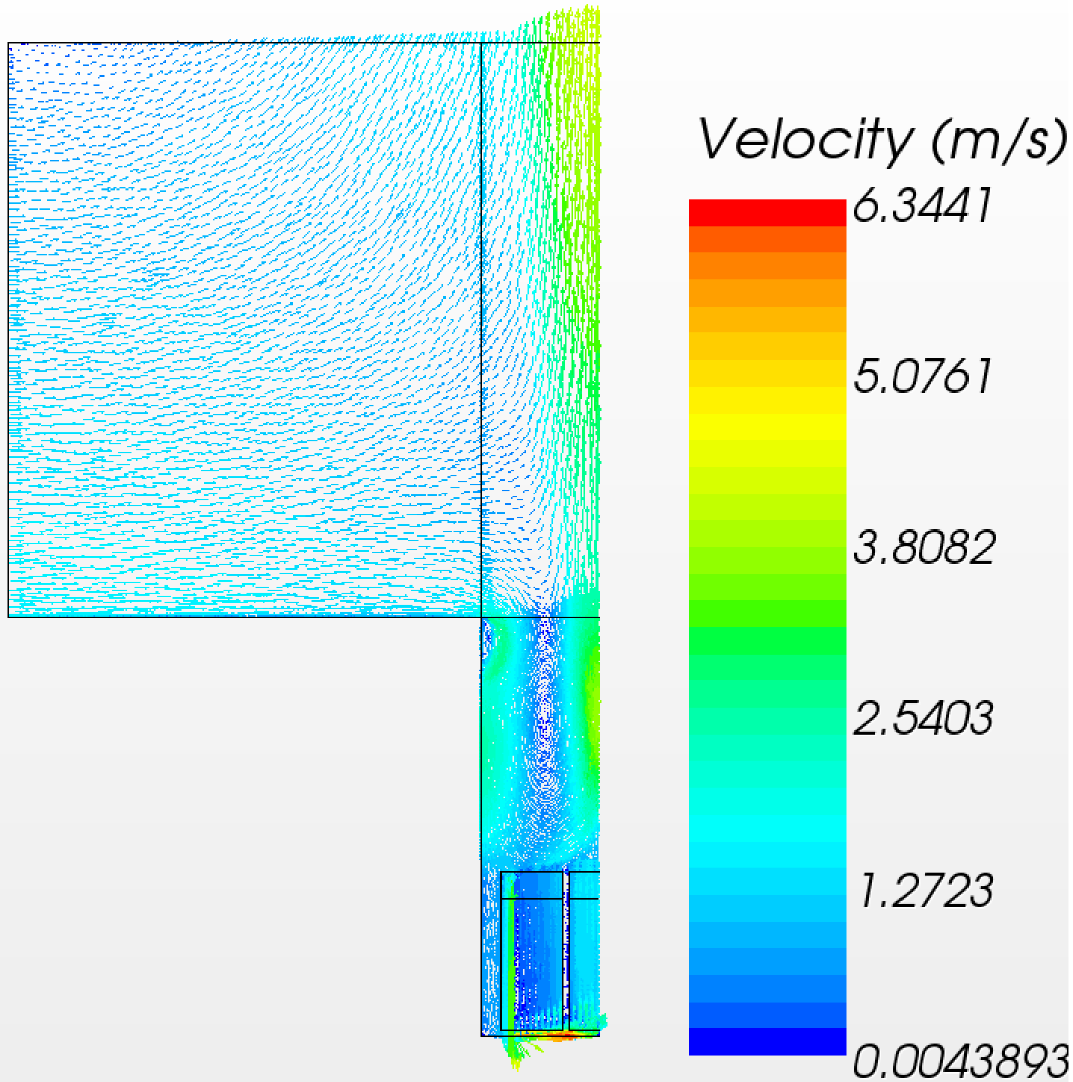
CFD評価モデル（２／２）



圧力境界
大気解放
（便宜上、建屋天井高さを設定）
解析領域
対称境界
使用済燃料貯蔵プール上端
（オペフロ高さ）
壁境界
使用済燃料ラック
使用済燃料貯蔵プール下端

CFD解析結果





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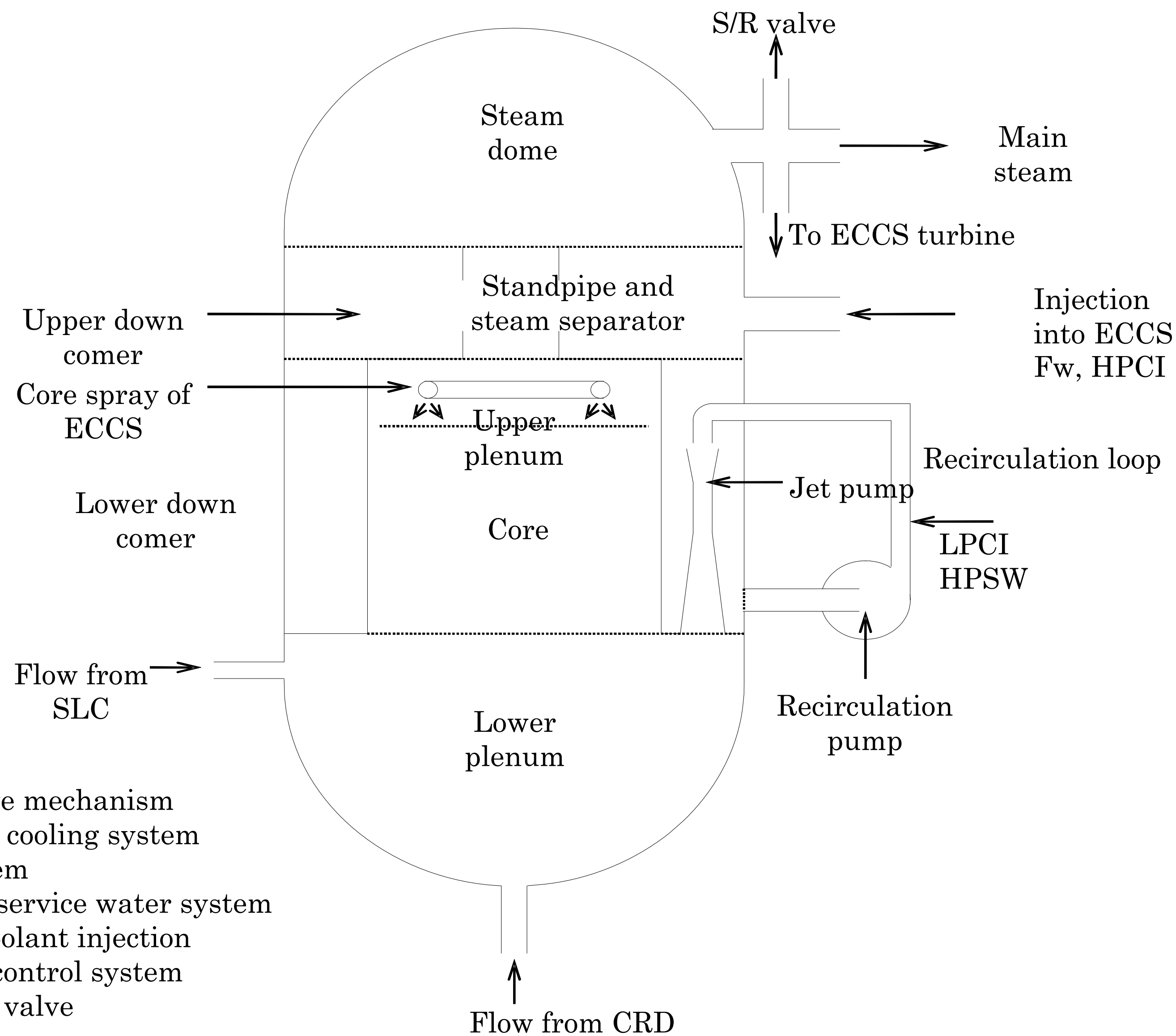
Plant conditions of Unit 1

Item	Conditions
Initial reactor output	1,380 MWt (rated output)
Initial reactor pressure	7.03 MPa [abs] (normal operating pressure)
Initial reactor water level	About 4,187 mm (normal water level: TAF standard)
Number of active core node split	Radial direction: 5 nodes Axial direction: 10 nodes
Burst temperature of cladding	Burst temperature of cladding tube727℃ (1,000 K)
Fuel melting	Each melting point of core component materials or average melting temperature of mixed materials
Spatial capacity of reactor containment vessel	D/W space: 3,410 m^3 S/C space: 2,620 m^3
Water volume in S/C	1,750 m^3
Decay heat	ANSI/ANS5.1-1979 model (Parameters are adjusted to obtain equivalent decay heat to the results of decay heat evaluation through ORIGEN2 by inputting fuel loading history.)

Plant conditions of Unit 2/3

Item	Conditions
Initial reactor output	2,381 MWt (rated output)
Initial reactor pressure	7.03 MPa [abs] (normal operating pressure)
Initial reactor water level	About 5,274 mm (normal water level: TAF standard)
Number of active core node split	Radial direction: 5 nodes Axial direction: 10 nodes
Burst temperature of cladding	Burst temperature of cladding tube727℃ (1,000 K)
Fuel melting	Each melting point of core component materials or average melting temperature of mixed materials
Spatial capacity of reactor containment vessel	D/W space: 4,240 m^3 S/C space: 3,160 m^3
Water volume in S/C	2,980 m^3
Decay heat	ANSI/ANS5.1-1979 model (Parameters are adjusted to obtain equivalent decay heat to the results of decay heat evaluation through ORIGEN2 by inputting fuel loading history.)

Outline of MAAP reactor pressure vessel model



CRD Control rod drive mechanism
ECCS Emergency core cooling system
FW Feedwater system
HPSW High pressure service water system
LPCI Low pressure coolant injection
SLC Standby liquid control system
S/R valve Safety relief valve

Time line of unit 1 (1/3)

Analysis Condition				Classification
No	Time and Date		Analyzed Events	
1	March 11th	2:46 PM	Earthquake occurred	○
2		2:46 PM	Reactor scram occurred	○
3		2:47 PM	MSIV closed	○
4		2:52 PM	IC(A) (B) activated automatically	○
5		Around 3:03 pm	IC(A) stopped	○
6		Around 3:03 pm	IC(B) stopped	○
		3:07 PM	Torus cooling with CCS (A) activated	○
		3:10 PM	Torus cooling with CCS (B) activated	○
		3:37 PM	Torus cooling with CCS (A)(B) stopped	○
7		3:17 PM	IC(A) restarted	△
8		3:19 PM	IC(A) stopped	△
9		3:24 PM	IC(A) restarted	△

Time line of unit 1 (2/3)

10		3:26 PM	IC(A) stopped	△
11		3:32 PM	IC(A) restarted	△
12		3:34 PM	IC(A) stopped	△
13		3:37 PM	SBO occurred	○
14		6:18 PM	IC(A) system 2A, 3A valve opened, and steam generation was confirmed	□
15		6:25 PM	IC(A) system 3A valve closed	□
16		8:50pm	The lineup of alternative water injection to the RPV with Diesel Driving Fire	□
17		9:30 PM	IC 3A valve was opened	□
19	May 12th	1:25am	D/D-FP stopped	□
		Around 4:00 AM	Batch injection of fresh water by the fire pump (1300 l of fresh water)	○
20		5:46 AM	Injection of fresh water by the fire pump started	○

Time line of unit 1 (3/3)

		9:05 AM	Start of drywell venting announced to press. (ignored)	○
21		2:30 PM	Regarding the containment vessel vent, operation of AO valve of suppression	△
22		2:53 PM	Injection of fresh water terminated	○
23		3:03 PM	Vent valve of PCV was closed	△
24		3:36 PM	Explosion of reactor building of Unit 1 occurred	○
25		7:04 PM	Injection of sea water started	○

Time line of unit 2 (1/3)

Analysis Condition				Classification
No	Time and Date		Analyzed Events	
1	March 11th	2:46 PM	Earthquake occurred	○
2		2:47 PM	Reactor scram occurred	○
		2:50 PM	RCIC activated manually	○
		2:51 PM	RCIC tripped (L-8)	○
3		3:02 PM	RCIC activated manually	○
		3:00 PM to 3:36 PM	RHR operation by S/C cooling mode	○
4		3:28 PM	RCIC tripped (L-8)	○
		3:39 PM	RCIC activated manually	○
5		3:41 PM	SBO occurred	○
6	March 12th	4:20 AM - 5:00 am	Changed the water source of RCIC from condensate storage tank to suppression chamber	○

Time line of unit 2 (2/3)

7	March 14th	1:25 PM	RCIC stopped (Official records)	○
8		4:34 PM	Started the operation of pressure reduction of Reactor Pressure Vessel （1 SRV open）	○
		4:34 PM	Started the injection of sea water through the fire protection system	○
9		Around 6:00 pm	Confirmed the decrease of the reactor pressure	○
10		7:20 PM	The fire pump stopped resulted from fuel run-out	○
11		7:54 PM	The fire pump activated	○
		7:57 PM	The second fire pump activated	○
12		9:20 PM	By opening 2 SRVs, reactor pressure decreased and water level recovered	○

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Time line of unit 2 (1/3)

13		Around 11:00 pm	It is presumed that the 1 SRV was closed	△
		11:25 PM	It is presumed that the 1 SRV was opened	△
	March 15th	1:10 AM	It is presumed that the 1 SRV was opened	△
		2:22 AM	It is presumed that the 1 SRV was opened	△
~~14~~	~~March 15th~~	~~Around 6:14 am~~	~~An abnormal sound has occurred near the suppression chamber, and the pressure inside decreased~~	~~○~~

Time line of unit 3 (1/5)

Analysis Condition				Classification
No	Time and Date		Analyzed Events	
1	March 11th	2:46 PM	Earthquake occurred	○
2		2:47 PM	Reactor scram occurred	○
3		3:05 PM	RCIC activated manually	○
4		3:25 PM	RCIC tripped （L-8）	○
5		3:38 PM	SBO occurred	○
6		4:03 PM	RCIC activated manually	○
7	March 12th	11:36 AM	RCIC tripped	○
		12:06 PM	S/C spray by DDFP started	○
8		12:35 PM	HPCI activated (L-2)	○
9	March 13th	2:42 AM	HPCI stopped	○
		3:05 AM	S/C spray by DDFP stopped	○
		5:08 AM	S/C spray by DDFP started	○

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Time line of unit 3 (2/5)

		7:39 AM	D/W spray by DDFP started	○
		7:43 AM	S/C spray by DDFP stopped	○
		8:40 AM - 9:10 am	D/W spray by DDFP stopped	○
10		Around 9:08 am	Started the pressure decrease of reactor pressure vessel by operating the SRV	○
11		Around 9:20 am	Regarding the PCV vent, pressure decrease was confirmed	○
12		9:25 am	Started the injection of fresh water	○
13		11:17 am	Regarding the PCV vent, a closure of AO valve of vent line caused by the slip out of driving air pressure was confirmed	○
		0:20 pm	Injection of fresh water was halted in order to depletion of water source	○
14		0:30 pm	Regarding the PCV vent, opening operation was implemented	○

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Time line of unit 3 (3/5)

15		1:12 pm	Injection of water was changed from fresh water to sea water	○
16		2:10 pm	Regarding the PCV vent, it is presumed that the vent valve is closed	△
		9:10 pm	S/C vent open (Reduction of D/W pressure was	
	March 14th	0:50 am	S/C vent close	
17	March 14th	1:10 am	Injection of water was halted in order to supply water to water source pit	○
18		3:20 am	The supply to the water source pit was finished and the injection of water	○
19		6:10 am	Regarding the PCV vent, AO valve of suppression chamber side was opened.	○
		11:01 am	Injection of water was halted in order to hydrogen explosion	
20		0:00 pm	Regarding the PCV vent, it is presumed that the valve of suppression chamber	△

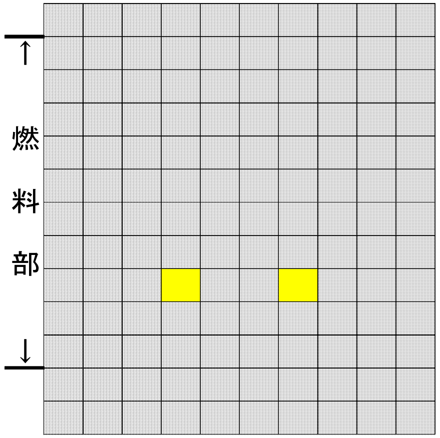
Time line of unit 3 (4/5)

21		4:00 pm	Regarding the PCV vent, it is presumed that the valve of suppression chamber	△
		4:30 pm	injection of water restarted	
22		9:04 pm	Regarding the PCV vent, it is presumed that the valve of suppression chamber	△
23	March 15th	4:05 pm	Regarding the PCV vent, the valve of suppression chamber side was opened.	○
24	March 16th	1:55 am	Regarding the PCV vent, the valve of suppression chamber side was opened.	△
25	March 17th	9:00 pm	Regarding the PCV vent, closure of the valve of suppression chamber side was confirmed.	△
26		9:30 pm	Regarding the PCV vent, the valve of suppression chamber side was opened.	△
27	March 18th	5:30 am	Regarding the PCV vent, closure of the valve of suppression chamber side was confirmed.	—

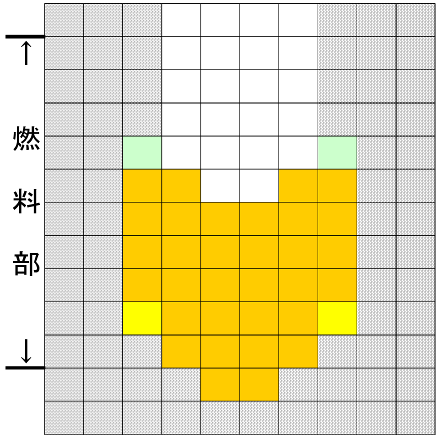
Time line of unit 3 (5/5)

28		Around 5:30 am	Regarding the PCV vent, the valve of suppression chamber side was opened.	—
29	March 19th	11:30 am	Regarding the PCV vent, closure of the valve of suppression chamber side was confirmed.	—
30	March 20th	Around 11:25 am	Regarding the PCV vent, the valve of suppression chamber side was opened.	—

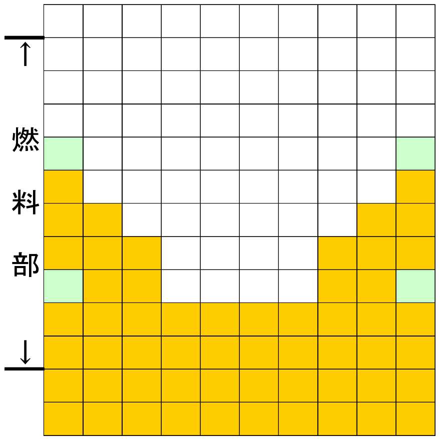
Unit 1 Analysis Results – Core Conditions



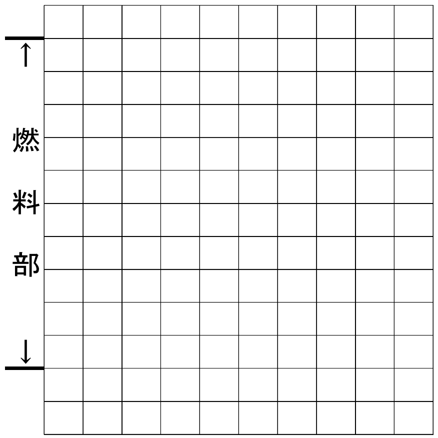
Approx. 4.8 hours after scram



Approx. 5.5 hours after scram



Approx. 7.6 hours after scram

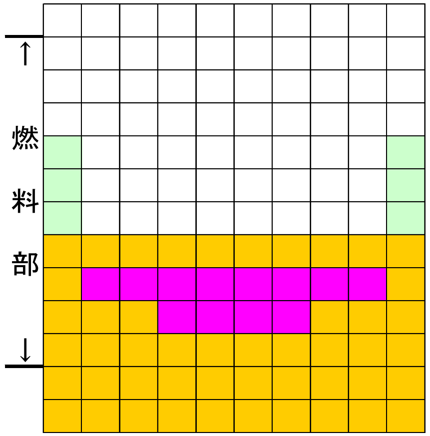


Approx. 8.6 hours after scram

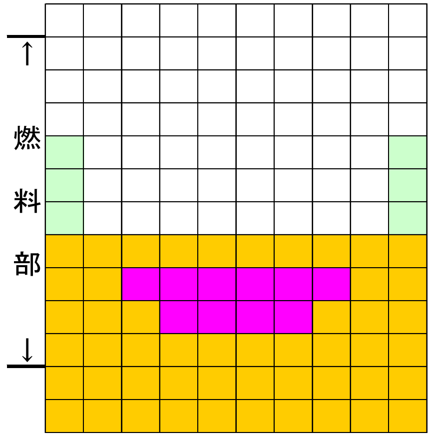
Model of damage states

- (white): No fuel (collapse)
- (gray): Normal fuel
- (light green): Damaged fuel accumulates (fuel rod form maintained)
- (yellow): Melted fuel flows down the cladding surface and cools and solidifies on the control rod surface, increasing the diameter of the control rods
- (orange): Fuel rod diameter further increases and flow channel is blocked by fuel
- (magenta): Melt pool formed

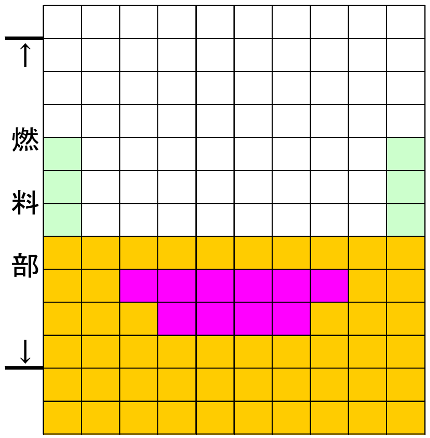
Unit 2 Analysis Results – Core Conditions



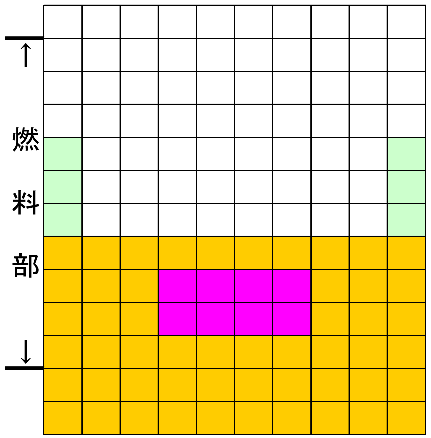
Approx. 96 hours after scram



Approx. 120 hours after scram



Approx. 144 hours after scram

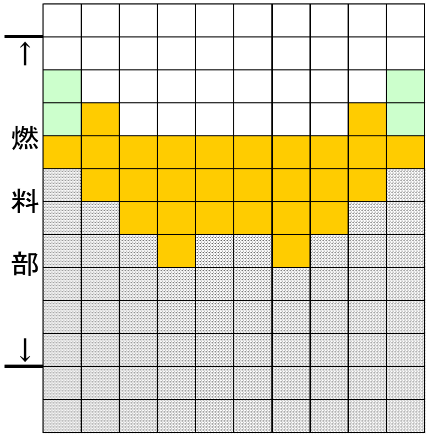


Approx. 168 hours after scram

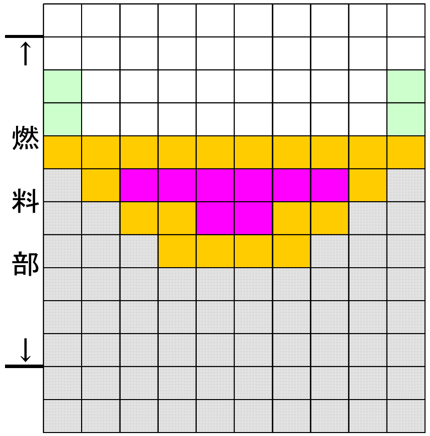
Model of damage states

- (white) : No fuel (collapse)
- (grey) : Normal fuel
- (light green) : Damaged fuel accumulates (fuel rod form maintained)
- (yellow) : Melted fuel flows down the cladding surface and cools and solidifies on the control rod surface, increasing the diameter of the control rods
- (orange) : Fuel rod diameter further increases and flow channel is blocked by fuel
- (magenta) : Melt pool formed

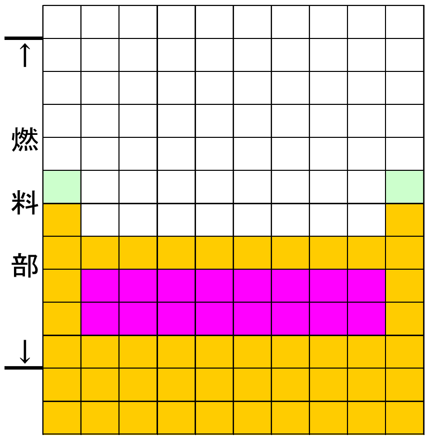
Unit 3 Analysis Results – Core Conditions



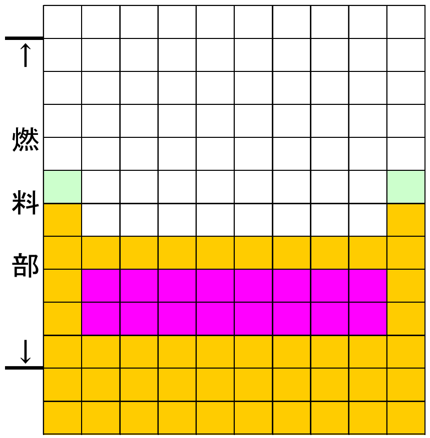
Approx. 48 hours after scram



Approx. 72 hours after scram



Approx. 96 hours after scram



Approx. 168 hours after scram

Model of damage states

- (white): No fuel (collapse)
- (grey): Normal fuel
- (light green): Damaged fuel accumulates (fuel rod form maintained)
- (yellow): Melted fuel flows down the cladding surface and cools and solidifies on the control rod surface, increasing the diameter of the control rods
- (orange): Fuel rod diameter further increases and flow channel is blocked by fuel
- (magenta): Melt pool formed

Who decided the analytical condition for rod failure when PCT is above 1000K?

- As you know the Van Houten report(NUREG0562) or Post-BT standard by AESJ*, the analytical condition of fuel failure when PCT become above 1000K is conservative.
- Furthermore, dry tube failure at the same condition is too conservative.
- We employed that analytical condition as just a typical case of dry tube failure.
- In fact, in case of 1F1, the temperature rising rate at 1000K is about 100K/min.
- Therefore, the difference in failure criterion for PCT does NOT make much of a difference.

*: S. Mizokami et.al., "Application of the Post-Boiling-Transition Standard to Licensing Analysis", Nuclear Technology, Volume 152, Number 1, 105-117, October 2005