MITSUBISHI HEAVY INDUSTRIES, LTD.
16-5, KONAN 2-CHOME, MINATO-KU

December 19, 2008

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. Jeffery A. Ciocco,

Docket No. 52-021
MHI Ref: UAP-HF-08278

Subject: MHI's Response to US-APWR DCD RAI No. 107

Reference: 1) "Request for Additional Information No. 107-1293 Revision 0, SRP Section: 03.09.04 – Control Rod Drive Systems, Application Section: 3.9.4," dated 11/24/2008

With this letter, Mitsubishi Heavy Industries, Ltd. ("MHI") transmits to the U.S. Nuclear Regulatory Commission ("NRC") documents as listed in Enclosure.

Enclosed is the response to 1 RAI contained within Reference 1.

As indicated in the enclosed materials, this submittal contains information that MHI considers proprietary, and therefore should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential. A non-proprietary version of the document is also being submitted with the information identified as proprietary redacted and replaced by the designation "[ ]".

This letter includes a copy of the proprietary version (Enclosure 2), a copy of the non-proprietary version (Enclosure 3), and the Affidavit of Yoshiki Ogata (Enclosure 1) which identifies the reasons MHI respectfully requests that all materials designated as "Proprietary" in Enclosure 2 be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

Please contact Dr. C. Keith Paulson, Senior Technical Manager, Mitsubishi Nuclear Energy Systems, Inc. if the NRC has questions concerning any aspect of the submittals. His contact information is provided below.

Sincerely,

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

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NRO

Enclosures:

1. Affidavit of Yoshiki Ogata

2. "Response to Request for Additional Information No. 107-1293, Revision 0" (Proprietary Version)

3. "Response to Request for Additional Information No. 107-1293, Revision 0" (Non-Proprietary Version)

4. Reference-1 "Improvement of CRDM Durability for PWR Plants, The Part of Mitsubishi Nuclear Technical Report No.54,1989"

CC: J. A. Ciocco
C. K. Paulson

Contact Information
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Enclosure 1

Docket No. 52-021
MHI Ref: UAP-HF-08278

MITSUBISHI HEAVY INDUSTRIES, LTD.

AFFIDAVIT

I, Yoshiki Ogata, state as follows:

1. I am General Manager, APWR Promoting Department, of Mitsubishi Heavy Industries, LTD ("MHI"), and have been delegated the function of reviewing MHI's US-APWR documentation to determine whether it contains information that should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4) as trade secrets and commercial or financial information which is privileged or confidential.

2. In accordance with my responsibilities, I have reviewed the enclosed document entitled "Response to Request for Additional Information No. 107-1293, Revision 0", dated December 19, 2008, and have determined that portions of the document contain proprietary information that should be withheld from public disclosure. Those pages contain proprietary information are identified with the label "Proprietary" on the top of the page, and the proprietary information has been bracketed with an open and closed bracket as shown here "[ ]". The first page of the document indicates that all information identified as "Proprietary" should be withheld from public disclosure pursuant to 10 C.F.R. § 2.390 (a)(4).

3. The information identified as proprietary in the enclosed document has in the past been, and will continue to be, held in confidence by MHI and its disclosure outside the company is limited to regulatory bodies, customers and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and is always subject to suitable measures to protect it from unauthorized use or disclosure.

4. The basis for holding the referenced information confidential is that it describes the unique design parameters developed by MHI for the Control Rod Drive Mechanism.

5. The referenced information is being furnished to the Nuclear Regulatory Commission ("NRC") in confidence and solely for the purpose of information to the NRC staff.

6. The referenced information is not available in public sources and could not be gathered readily from other publicly available information. Other than through the provisions in paragraph 3 above, MHI knows of no way the information could be lawfully acquired by organizations or individuals outside of MHI.

7. Public disclosure of the referenced information would assist competitors of MHI in their design of new nuclear power plants without incurring the costs or risks associated with the design of the subject systems. Therefore, disclosure of the

information contained in the referenced document would have the following negative impacts on the competitive position of MHI in the U.S. nuclear plant market:

A. Loss of competitive advantage due to the costs associated with the development of the unique design parameters.

B. Loss of competitive advantage of the US-APWR created by the benefits of the Control Rod Drive Mechanism operation.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information and belief.

Executed on this 19th day of December 2008.

Y. Ogata

Yoshiki Ogata,
General Manager- APWR Promoting Department
Mitsubishi Heavy Industries, LTD.

Enclosure 3

UAP-HF-08278
Docket No. 52-021

Response to Request for Additional Information No. 107-1293,
Revision 0

December 2008
(Non-Proprietary)

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION

12/19/2008

US-APWR Design Certification

Mitsubishi Heavy Industries

Docket No. 52-021

RAI NO.:	**NO. 107-1293 REVISION 0**
SRP SECTION:	**03.09.04 – CONTROL ROD DRIVE SYSTEMS**
APPLICATION SECTION:	**3.9.4**
DATE OF RAI ISSUE:	**11/24/08**

US-APWR Design Certification- 03.09.04, Control Rod Drive Systems (CRDS)
[Review performed against revision 0 of the US-APWR DCD Tier 2.]

QUESTION NO. : RAI 1293-01

Include reference(s) that documents control rod drive mechanism (CRDM) qualification to operate in the reactor pressure vessel (RPV) environment. Based on the nature of this reference, provide one of the following for review:

1. For a new series of tests unique to the US-APWR CRDM, provide for review an operability assurance program for the US-APWR CRDM that covers all the items contained in the guidance in SRP Section 3.9.4, Part I, Item 4, or

2. If a specific previous testing program that has been approved by the USNRC is referenced, such as for the L-106A CRDM, provide the following additional information for review:

a. Describe differences between the US-APWR and the previous design, such as the L-106A CRDM and discuss their effects on the applicability of the previous operability tests to the US-APWR CRDM. US-APWR DCD Tier 2, Section 3.9.4.1.1 (page 3.9-56) states that the US-APWR CRDM design is improved by (1) butt welding the CRDM latch housing to the CRDM nozzle on the reactor vessel closure head and (2) applying a chrome carbide coating to the latch arms.
b. Identify any differences in the operating conditions, such as the weight of the rod control cluster assembly (RCCA) and loads imposed by hydrodynamic forces through the RCCA to the CRDM, and discuss their effects on the applicability of the previous tests to the US-APWR CRDS. US-APWR DCD Tier 2, Section 1.2.1.5.1.1 (page 1.2-11) states that the active fuel length of the US-APWR will be increased from 12 to 14 ft as compared to the current Mitsubishi-APWR design.

Therefore, the rod control cluster assembly (RCCA) of the US-APWR may be heavier than in previous designs, and the increased weight may affect functionality and wear differently than in previous tests. US-APWR DCD Tier 2, Section 1.5.2.1 (page 1.5-1) indicates that there are changes in the reactor internals which may alter flow loads from those in previous designs. US-APWR DCD Tier 2, Section 4.3.4 (Page 4.3-27) states there the number of fuel assemblies has been increased to 257 from previous designs.
c. Compare the design LOCA plus SSE loads for the US-APWR CRDS to the loads that were used in the previous design verification tests.
d. Provide the basis for the 60-year lifetime for the CRDM internals. The design lifetime for the L-106A CRDM was 40 years. US-APWR DCD Tier 2, Section 3.9.4.2.1 (Page 3.9-60) states that the design life for the US-APWR CRDM is 60 years.

General Design Criteria (GDC) 2, 26, 27, and 29, require that the CRDS be designed to withstand the effects of an earthquake, and be designed with appropriate margin to assure its functionality under conditions of normal operation, anticipated operational occurrences, and the postulated accident conditions. The guidance in USNRC Standard Review Plan (SRP), Section 3.9.4, Part I **AREAS OF REVIEW,** Item 4 (page 3.9.4-3) states that a review of plans for the conduct of an operability assurance program or that references previous test programs or standard industry procedures for similar apparatus is performed. The guidance in USNRC Standard Review Plan (SRP), Section 3.9.4, Part III **REVIEW Procedures,** Item 1 (page 3.9.4-8) states that, "The objectives of the review are to determine...that suitable life cycle testing programs have been utilized to prove operability under service conditions".

ANSWER:

Review option 2 is applicable, as the design and operating conditions of the US-APWR CRDM are compared with that of the previously tested L-106A CRDM.

The CRDM for the US-APWR design is based on the L-106A type CRDM, which has been used in many operating plants in the USA and Japan. CRDMs for the US-APWR have incorporated two design improvements. As noted by question RAI 1293-01, the US-APWR design utilizes butt welding instead of a threaded connection and canopy seal weld to assure an extremely low probability of leakage, and chrome carbide coating is applied on tip of the latch arms where it engages the control rod drive rod to improve resistance to wear.

Butt welding design is used in both USA and Japanese plants, and chrome carbide coating is used in Japanese plants. Applicability of the US-APWR improved design features is described below. The structural integrity of the pressure housing under loss of coolant accident (LOCA) coincident with safe shutdown earthquake (SSE) conditions will be reported to the NRC by the end of March, 2009. After this action, the response to question RAI 1293-01, review option 2, part c will be provided in April, 2009.

a. Describe differences between the US-APWR and the previous design, such as the L-106A CRDM, and discuss their effects on the applicability of the previous operability tests to the US-APWR CRDM.

   (1) Latch Assembly

   The design of the US-APWR CRDM latch assembly is the same as the previous L-106A type CRDM, except the chrome carbide coating on the latch arms to improve wear resistance. The latch assembly is shown in Figure 1. This design change does not affect operability, since the thickness of the chrome carbide coating is thin enough compared with the clearance between the latch tip and the groove of the drive rod.

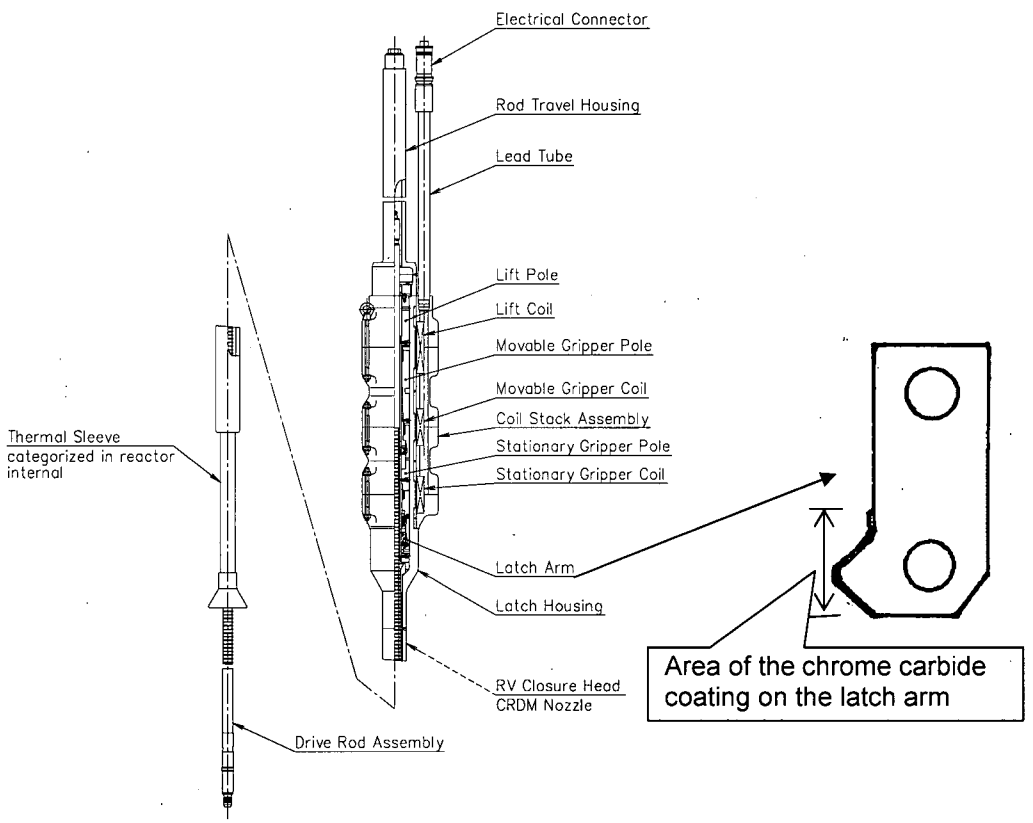


Figure 1 Applied Area of the Chrome Carbide Coating

(2) Pressure Housing

The US-APWR CRDM and previous CRDM pressure housings are shown in Figure 2 to illustrate three areas of design improvements for the cap, rod travel housing and latch housing connections. The previous CRDM design connections of the cap, rod travel housing and latch housing are threaded and canopy seal welded. For the US-APWR CRDM pressure housing, the cap and rod travel housing is machined from one piece of material, and the latch housing and CRDM adapter is also machined from one piece of material. The pressure housing consists of a rod travel housing and a latch housing, both of which are butt welded. The latch housing is butt welded to a CRDM nozzle of the reactor vessel (RV) head. This improved design results in an extremely low probability of primary coolant system leakage. This design change does not affect operability, because interface condition between the latch housing and the latch mechanism is not changed.

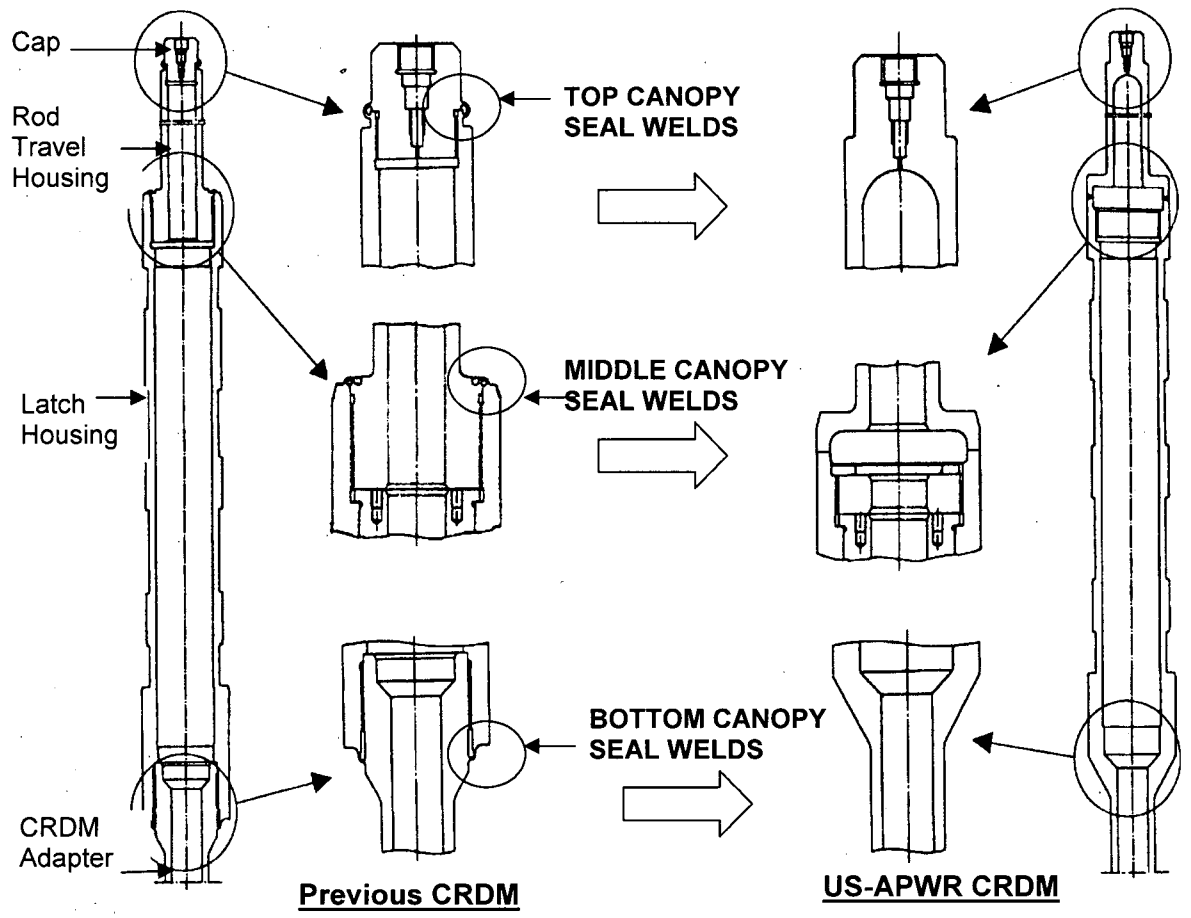


Figure 2 Comparison Sketch Between Previous CRDM and US-APWR CRDM

(3) Drive Rod Assembly

The length of the US-APWR drive rod is extended from the previous design to accommodate the 14-ft fuel and the larger size of the RV. The weight of the drive rod is increased, but operability is maintained, because the effect is small enough. Please see answer b (3) and d.

(4) Coil Stack Assembly

The US-APWR coil stack assembly design is same as previous design.

b. Differences and discussion of effects on operability

(1) Without canopy seal design of the pressure housing

The interface between the latch mechanism and the latch housing is not changed. Therefore this design change does not affect operability.

(2) Chrome carbide coating on the tip of the latch arm

Thickness of the chrome carbide coating on the latch arm is [ ] mils, which is thin enough compared with the approximately [ ] inch clearance between the latch tip and groove of the drive rod. Therefore, chrome carbide coating does not affect operability.

(3) Extension of the drive rod

Drive line weight of the US-APWR is [ ] lbs ([ ] kg), which is about a 10 % increase from current 4 loop drive line weight. This weight increase is within the tested capability of the CRDM noted in Reference 1.

(4) Plant operating conditions

Plant operating conditions are described in DCD Table 4.4-1.

Pressure of the primary coolant water of US-APWR is the same as current 4 loop plants. Temperature and core average coolant velocity of US-APWR are slightly lower than current 4 loop plants.

Hydrodynamic forces in the reactor are upward forces, such as a flow force in the core region, which act in opposite direction of the drive line weight. The verification test (Reference 1) was conducted in high pressure and high temperature water, however the core flow is not simulated in order to provide a conservative test. The effect of decreasing core average coolant velocity is covered by the verification test.

c. Effect of LOCA and SSE loads

Effect of LOCA and SSE loads is verified by comparison between the estimated deflection of the pressure housing and the allowable limit described in DCD Tier 2, Subsection 3.9.4.3. Confirmation that the estimated deflection is within the allowable limit will be provided after the stress report of pressure housing is submitted to the NRC in March, 2009.

d. The bases for assuring the 60 year design life for the US-APWR CRDM are described below.

(1) The stress and fatigue strength of the pressure boundary is evaluated by application of design transients covering 60 years of expected plant life. The results are scheduled to be reported to the NRC in March, 2009.

(2) The integrity of CRDM latch mechanism was confirmed by the endurance test (Reference1). The test was conducted by using a 12 foot drive line CRDM with chrome carbide coating. The drive line weight used in this test was 324 lbs (147 kg), which is slightly heavier than the US-APWR drive line weight of [ ] lbs ([ ] kg), and the functionality was confirmed to ten million steps. The required step numbers of a 40 year life time is two and one-half million steps. Therefore, a ten million step endurance test is conservatively bound to be enough for the 60 year design life time (Reference 1).

(3) The coil stack assembly and drive rod assembly are not required to have a 60 year operating time. These assemblies can be replaced during the life of the plant.

Reference 1: Improvement of CRDM Durability for PWR Plant, Mitsubishi Nuclear Technology Report No. 54, 1989.

Impact on DCD

DCD Revision 2 will incorporate the following change:

- Insert as the last sentence in the first paragraph in Subsection 3.9.4.1.1: "These design improvements do not affect operability."

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

QUESTION NO. : RAI 1293-02

Provide for review how wear and overcoming a stuck rod are addressed in the operability assurance program, including details of the improved wear resistance offered by the chrome carbide coating that has not previously been used in U.S. nuclear power plants.

General Design Criteria (GDC) 26, 27, and 29, as they relate to the CRDS, require that the CRDS be designed with appropriate margin to assure its functionality under conditions of normal operation, postulated accident conditions, and anticipated operational occurrences. The guidance in USNRC Standard Review Plan (SRP), Section 3.9.4, Part II ***SRP Acceptance Criteria***, Item 4 (page 3.9.4-6) states that, "The operability assurance program will be acceptable provided that observed performance as to wear, functioning times, latching, and ability to overcome a stuck rod meet system design requirements." US-APWR DCD Tier 2, Section 3.9.4.4 (page 3.9-62) states that, "The capability of the CRDM functions, including withdrawal, insertion, and trip delay are confirmed by both lead unit tests and production unit tests to demonstrate that the design specification requirements are met prior to shipment." System design requirements for cold stepping, hot and cold trip delay times, and hot stepping are given in US-APWR DCD Tier 2 Section 3.9.4, and preoperational tests are discussed in Section 14.2, but there is no discussion on wear or overcoming a stuck rod.

ANSWER:

The operability assurance to limit wear is achieved by the chrome carbide coating of the latch arms. The need to overcome a stuck rod is avoided by the design as discussed below.

(1) The critical wear of the latch mechanism occurs at the surface of latch arms where it contacts with the drive rod. Chrome carbide coating is applied on the tip of the latch arms. The wear and reliability of the improved CRDM was confirmed by the test as discussed in Reference 1 of RAI 1293-01.

(2) The clearances in the US-APWR CRDM latch assembly, each part are designed to avoid a stuck rod condition. The thermal expansion of each part is evaluated to determine the clearances. This design is the same as L-106A, which reflects operationally-proven design and experience.

Impact on DCD

DCD Revision 2 will incorporate the following change:

- Insert as the last paragraph of Subsection 3.9.4.3, "The clearances in the CRDM latch assembly, the latch arm, and the coil assembly are controlled to avoid a stuck rod condition. The thermal expansion of each part is evaluated to determine and ensure the clearances. This design is the same as L-106A, which reflects operationally-proven design and experience."

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

QUESTION NO. : RAI 1293-03

Include the acceptance criteria for the safety-related non-pressurized portion of the USAPWR. Provide for review a description and results of stress, deflection, and fatigue analyses for the non-pressurized portion of the US-APWR CRDM, including the following:

- What are the design loads and loading combinations?
- What values of material properties are used and what is the justification for their basis?
- What stress, deflection, and fatigue criteria are used and what is the justification for their basis?
- What are the design margins and how do they compare with previous designs?

General Design Criteria (GDC) 2, 26, 27, and 29, as they relate to the CRDS, require that the CRDS be designed to withstand the effects of an earthquake, and be designed with appropriate margin to assure its functionality under conditions of normal operation, anticipated operational occurrences, and postulated accident conditions. The guidance in USNRC Standard Review Plan (SRP), Section 3.9.4, Part II ***SRP Acceptance Criteria***, Item 2.C (page 3.9.4-6) states that for "non-pressurized equipment (Non-ASME Code): Design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. A justification of any decreases in design margins should be provided." The guidance in USNRC Standard Review Plan (SRP), Section 3.9.4, Part II ***SRP Acceptance Criteria***, Item 3 (page 3.9.4-6) states that, "The stress limits applicable to...non-pressurized portions of the control rod drive system should be as given in SRP Section 3.9.3 for the response to each loading set." US-APWR DCD Tier 2, Section 3.9.4.2.3 (page 2.9-61) states that the non-pressurized portion of the USAPWR CRDM is non-ASME Code, Section III, limited; however, no description is provided on the criteria for structural analyses, design margins, or how design margins were obtained.

ANSWER:

The ASME Code requirements do not apply to non-pressurized components such as latch mechanism, the drive rod and the coil assembly. These non-pressurized components are classified as non-safety components. This is based on a file safe design with scram principle utilizing gravity. If the coil assembly or electric device of the CRDM fails, the control rods are dropped/inserted into the core by gravity and reduce the reactivity. If the drive rod fails, the control rods drop into the core and reduce the reactivity.

Design endurance criteria of the latch mechanism is six million steps, which accommodates a margin for 60 years of operation (e.g., maintain stepping function). The verification test result is provided in Reference 1 in reply for RAI 1293-01.

Impact on DCD

DCD Revision 2 will incorporate the following changes:

- Replace the 1st sentence of 2nd paragraph in Subsection 3.9.4.2.3 with the following: "The ASME Code requirements do not apply to non-pressurized components such as latch mechanism, the drive rod and the coil assembly. These non-pressurized components are classified as non-safety components. This is based on a file safe design with scram principle utilizing gravity. If the coil assembly or electric device of the CRDM fails, the control rods are dropped/inserted into the core by gravity and reduce the reactivity. If the

drive rod fails, the control rods drop into the core and reduce the reactivity."

- Insert as the last bullet in the 1st paragraph in Subsection 3.9.4.2.1: "Design endurance criterion of the latch mechanism is six million steps, accommodating a margin for 60 years of operation (e.g., maintain stepping function)."

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

QUESTION NO. : RAI 1293-04

Include design basis pipe breaks (DBPB) in the ASME Code Service Level C Design Load Combinations in Section 3.9.3 or provide for review a discussion as to why DBPB is not included in the ASME Code Service Level C Design Load Combinations.

General Design Criterion (GDC) 27, as it relates to the CRDS, requires that the CRDS be designed with appropriate margin to assure its functionality under accident conditions. The guidance in USNRC Standard Review Plan (SRP), Section 3.9.3, Appendix A, Table 1 (page 3.9.3-23) includes DBPB in both Emergency and Faulted System Operating Conditions (ASME Code Service Stress Limits C and D, respectively). Similar guidance is also found in USNRC Standard Review Plan (SRP), Section 3.9.3, Appendix A, Paragraph 4.B(iii)(1) (page 3.9.3-20). However, in US-APWR DCD Tier 2, Section 3.9.3, Table 3.9.3 (page 3.9-91), DBPB is listed in the ASME Code Service Level D Design Load Combinations, but not in the ASME Code Service Level C Design Load Combinations.

ANSWER:

RAI 1293-04 requested justification for not including design base pipe break (DBPB) loading in the Service Level C loading combinations, since DBPB loading is included in the Level C loadings suggested by SRP Section 3.9.3. The Service Level C pipe break is defined to be a maximum of 1 in. diameter pipe size in a Class 1 branch line small break LOCA. This is somewhat larger than the DBPB identified in SRP Section 3.9.3 Appendix A, which is equivalent to a 3/8th in. diameter break (i.e., the break size in a Class 1 branch line that results in the loss of reactor coolant at a rate less than or equal to the capability of the reactor coolant makeup system). Postulated breaks in 1 in. diameter piping and smaller piping, in accordance with guidance in SRP 3.6.2, do not require the analysis of the dynamic mechanical loadings from the ruptured pipe on reactor coolant system components and therefore are not included in US-APWR DCD Tier 2, Section 3.9.3, Table 3.9-3, which gives the loading combinations for mechanical loads.

A break in a 1 in. diameter Class 1 branch line results in reactor coolant system temperature and pressure transient conditions, which are included in the reactor coolant system design transients noted in DCD Table 3.9-1.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

QUESTION NO. : RAI 1293-05

Provide for review a description in more detail of the quality classification of the nonpressurized safety components of the CRDS (e.g., latch mechanism).

General Design Criterion (GDC) 1 and 10CFR50.55a, as they relate to the CRDS, require that the CRDS be designed to quality stands commensurate with the importance of the safety functions to be performed. US-APWR DCD Tier 2, Section 3.9.4.2.3 (page 3.9-61) states that, "The design, fabrication, inspection, and testing of the safety-related latch mechanism comes under the quality assurance requirement regarding safety components in 10CFR 50.55a..." However, non-pressurized safety component portions of the CRDM are not listed in US-APWR DCD Tier 2, Table 3.2.2 (pages 3.2-16 to 3.2-65), nor is a specific paragraph of 10CFR50.55a referenced, and the quality standards (e.g., such as NQA-1 or 10 CFR 50 Appendix B) to be applied <u>need to be clarified</u>.

ANSWER:

See response to RAI 1293-03.

The Quality Assurance Program Complies with ASME NQA-1 and 10 CFR 50.55a, Appendix B.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

QUESTION NO. : RAI 1293-06

Provide for review the basis of the 1.18-inch allowable rod travel housing deflection during the seismic event in US-APWR DCD Tier 2, Section 3.9.4.3 (page 3.9-62), and how it has been quantified by analysis that the rod control cluster assembly (RCCA) will be inserted into the core at this deflection.

General Design Criterion (GDC) 2, as it relates to the CRDS, requires that the CRDS be designed to withstand the effects of an earthquake. The guidance in USNRC Standard Review Plan (SRP), Section 3.9.4, Part I **AREAS OF REVIEW,** Item 1 (page 3.9.4-2) states that, "The descriptive information, including design criteria...is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly."

ANSWER:

The allowable rod travel housing deflection of 1.18 inches was determined by the rod insertion test. The maximum deflection of the rod travel housing due to the LOCA and seismic loads is obtained by the analysis.

The summary of test report is scheduled to be translated to English and to be submitted to the NRC in April, 2009, after the CRDM stress report is finalized in March, 2009.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

QUESTION NO. : RAI 1293-07

Include the criteria used for CRDM operational capability, including the margin, following exposure to the combined effects of a LOCA and an SSE.

General Design Criteria (GDC) 2 and 27, as they relate to the CRDS, require that the CRDS be designed to withstand the effects of an earthquake, and be designed with appropriate margin to assure its functionality under conditions of postulated accident conditions. The guidance in USNRC Standard Review Plan (SRP), Section 3.9.4, Part I **AREAS OF REVIEW,** Item 1 (page 3.9.4-2) states that, "The descriptive information, including design criteria, testing programs,...is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly."

ANSWER:

The CRDM safety function of scram capability is confirmed by limiting the maximum deflection of the rod travel housing due to a LOCA and SSE to less than 1.18 inch.

The design margin in response to this RAI will be provided in April, 2009 after the stress analysis is finished in March, 2009.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

QUESTION NO. : RAI 1293-08

Include a reference(s) that the CRDM design conforms to its design criteria and limits. If the design verification includes loading combination analysis in conjunction with testing, then include a reference(s).

General Design Criteria (GDC) 2 and 27, as they relate to the CRDS, require that the CRDS be designed to withstand the effects of an earthquake, and be designed with appropriate margin to assure its functionality under conditions of postulated accident conditions. The guidance in USNRC Standard Review Plan (SRP), Section 3.9.4, Part I **AREAS OF REVIEW,** Item 1 (page 3.9.4-2) states that, "The descriptive information, including design criteria, testing programs,...is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly."

ANSWER:

This stress report is identified in the DCD as Reference 3.9-59, Summary of Stress Analysis Results for Components and Piping, scheduled to be submitted to the NRC in March, 2009. As clarification, a sentence will be added in Subsection 3.9.4.4 during DCD Revision 2 to clarify the applicability of this reference to the CRDM design.

Impact on DCD

DCD Revision 2 will incorporate the following change:

- Insert as the last sentence in 2nd paragraph of Subsection 3.9.4.4: "Stress analysis results are provided in Reference 3.9-59."

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

QUESTION NO. : RAI 1293-09

Verify that the insertion and withdrawal times in the stepping mode, and the drop times, meet the design requirements. Provide the design requirements for these functions, their bases (for example, the safety analysis), and the margins between the CRDS functional requirement times and the times required by the safety analysis.

General Design Criteria (GDC) 26, 27, and 29, as they relate to the CRDS, require that the CRDS be designed with appropriate margin to assure its functionality under conditions of normal operation, postulated accident conditions, and anticipated operational occurrences (AOO). The guidance in USNRC Standard Review Plan (SRP), Section 3.9.4, Part I **AREAS OF REVIEW,** Item 1 (page 3.9.4-2) states that, "The descriptive information, including design criteria, testing programs,...is reviewed to permit an evaluation of the adequacy of the system to perform its mechanical function properly." Section 3.9.4.2.1 of US-APWR DCD Tier 2 (page 3.9-61), states that, "The rod drop time...is evaluated by analysis." However, no analysis is referenced and the type of analysis needs to be clarified.

ANSWER:

The design requirements from safety analysis during stepping mode are:

- Maximum speed: 72 steps/minute
- Scram delay time: within 0.15 second

Design requirements are the same as the functional requirements.

The functionality of the US-APWR CRDM is assured through many years of operating experience in Japan. Production tests are performed on all CRDMs prior to shipment to demonstrate that the design specification requirements are met.

The effect for the rod drop time is evaluated by the calculated deflection of the CRDM pressure housing and test result, which is basis of the deflection criteria of the CRDM pressure housing.

Impact on DCD

There is no impact on DCD.

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

QUESTION NO. : RAI 1293-10

Clarify if all CRDMs go through the functional verification tests, and at what stage (including post-refueling). Provide for review the test abstract for the Control Rod Drive System referred to in US-APWR DCD Tier 2 Section 14.3.4.7.

General Design Criteria (GDC) 26, 27, and 29, as they relate to the CRDS, require that the CRDS be designed with appropriate margin to assure its functionality under conditions of normal operation, postulated accident conditions, and anticipated operational occurrences. Section 4.6.3 in US-APWR DCD Tier 2 (page 4.6-2) lists four stages of tests: prototype tests of components, production tests of components following manufacture in shop, preoperational tests on site, and periodic in-service tests, which are stated to be in Section 3.9.4.4 and Section 14.2. These Sections give some information on preshipment and preoperational testing, but none on periodic in-service or post-refueling startup tests. The tests included in each stage need to be clarified, and whether each CRDM must be tested. Section 14.3.4.7 in US-APWR DCD Tier 2 (page 14-16) refers to Section 14.2.9.1.8 for a test abstract on Control Rod Drive Systems, but this section is not included in the Tier 2 information.

ANSWER:

Functional testing is performed in the shop on all CRDMs before shipping to the plant site. At the plant site, all CRDMs are tested to confirm the functionality as described in Subsection 14.2.12.2.1.5, Rod Drop Time Measurement Test, and Subsection 14.2.12.2.1.6, CRDM Operational Test. The stepping and the drop tests to be performed as in-service/post-refueling tests will be added in Subsection 3.9.4.4 in the next revision of the DCD. A sentence will also be added to Subsection 3.9.4.4 to clarify that all CRDMs go through the functional verification tests.

Impact on DCD

DCD Revision 2 will incorporate the following change:

- Insert at the end of Subsection 3.9.4.4:

"Post-Refueling Startup Test

- The stepping and the drop tests are performed as in-service/post-refueling tests. The criteria of this test are applicable to all CRDMs as described in Subsection 14.2."

Impact on COLA

There is no impact on COLA.

Impact on PRA

There is no impact on PRA.

Reference 1

Improvement of CRDM Durability for PWR Plants

The Part of Mitsubishi Nuclear Technical Report No.54,1989

Notes: This document is translated the part of Mitsubishi Nuclear Technical Report No.54, 1989.

1. Introduction

The L-106A Control Rod Drive Mechanism (hereafter, CRDM) is attached to upper part of the reactor vessel head. The drive rod of the CRDM connects with a control rod cluster (hereafter, control rods) in the reactor vessel. The CRDM withdraws, inserts, or holds the control rods to control the thermal power output. The drive speed can be adjusted within a range of 10 cm per minute (4 in/min) to 114 cm per minute (45 in/min) in normal operation. The CRDM provides high level safety as a fail-safe mechanism. In an emergency, electrical power for the CRDM is cut off, then the control rods are inserted in the core by their weight to shutdown the reactor.

For recent plants various types of CRDM operation such as frequency controlled operation and high-performance load follow operation have been requested, and improvement in the CRDM's durability is required. To improve this durability, at Mitsubishi Heavy Industries (here after, MHI) has developed an improved CRDM latch arm (hereafter, latch) whose tip is coated chromium carbide by thermal spraying. MHI carried out 10 million steps endurance test for this latch under high-temperature and high pressure water conditions that simulated the operating environment of an actual plant. MHI showed that it could endure use of more than 6 million steps under a variety of operating conditions.

2. Outline of the CRDM design

MHI has shown a cutaway view of Mitsubishi's standard CDRM in Figure-1, and we give an outline of its equipment specifications in Table-1. The CDRM is composed of four main parts described below—the latch assembly, the pressure housing, the drive rod assembly, and the coil assembly. As shown in Figure-2, the control rods are inserted or withdrawn by a method in which three kinds of operating coils become sequentially energized or de-energized. This type of CRDM drives the internal mechanism by electromagnetic power coming from the outside of the pressure housing. Thus it does not need penetration on the pressure housing, and there are no adverse effects on the pressure boundary's reliability.

The latch assembly is a mechanism that holds the drive rod and inserts or withdraws it in steps. Figure-3 shows an outline of this mechanism. The latch assembly consists of three groups of magnetic poles and plungers and two groups of latches (with three latches per group). The latch assembly moves the drive rod about 16 mm up and down per step by the series of actions shown in Figure-2. The latch assembly is installed in the pressure housing, and its moving parts are constantly lubricated by coolant while operating.

The pressure housing is consisted the latch housing and rod travel housing. The latch housing contains the latch assembly and on its outside supports the coil stack assembly. The rod travel housing makes space to withdrawn the drive rod in the pressure boundary.

The drive rod is a tube form product which outer diameter is about 45 mm(1.75 in). On its outer surface there are 16 mm pitched grooves that are latched by the latch arms. At the drive rod's lower end there is a coupling that connects with the control rods. Inside the drive rod there is a detachable rod for connecting with and disconnecting from the control rods. When the reactor vessel head is removed with CRDMs, the drive rods stay in the reactor, and it is possible to connect or disconnect the control rods and the drive rod by manipulating the detachable rod.

The coil stack assembly is set onto the outer circumference of the latch housing. A coil stack assembly contains three coils that operate in response to the latch assembly and are energized and de-energized by a controlled electric current.

Table — 1 CRDM Equipment Specifications

Quantity (Number needed for thermal output) control	
2 loop plant	29
3 loop plant	48
4 loop plant	53
Driving Method	
Normal Operation	Magnetically operated jacking type
Plant Trip	Insert by gravity
Speed	
Withdrawal and Insertion in normal operation	From about 10 cm / min (4 in / min) to 114 cm / min (45 in / min)
Rod drop time (Insert 85 % of full stroke)	About 2 sec
Design pressure	175kg/cm^{2}g (2500 psia)
Design temperature	343 degrees C (650 degrees F)

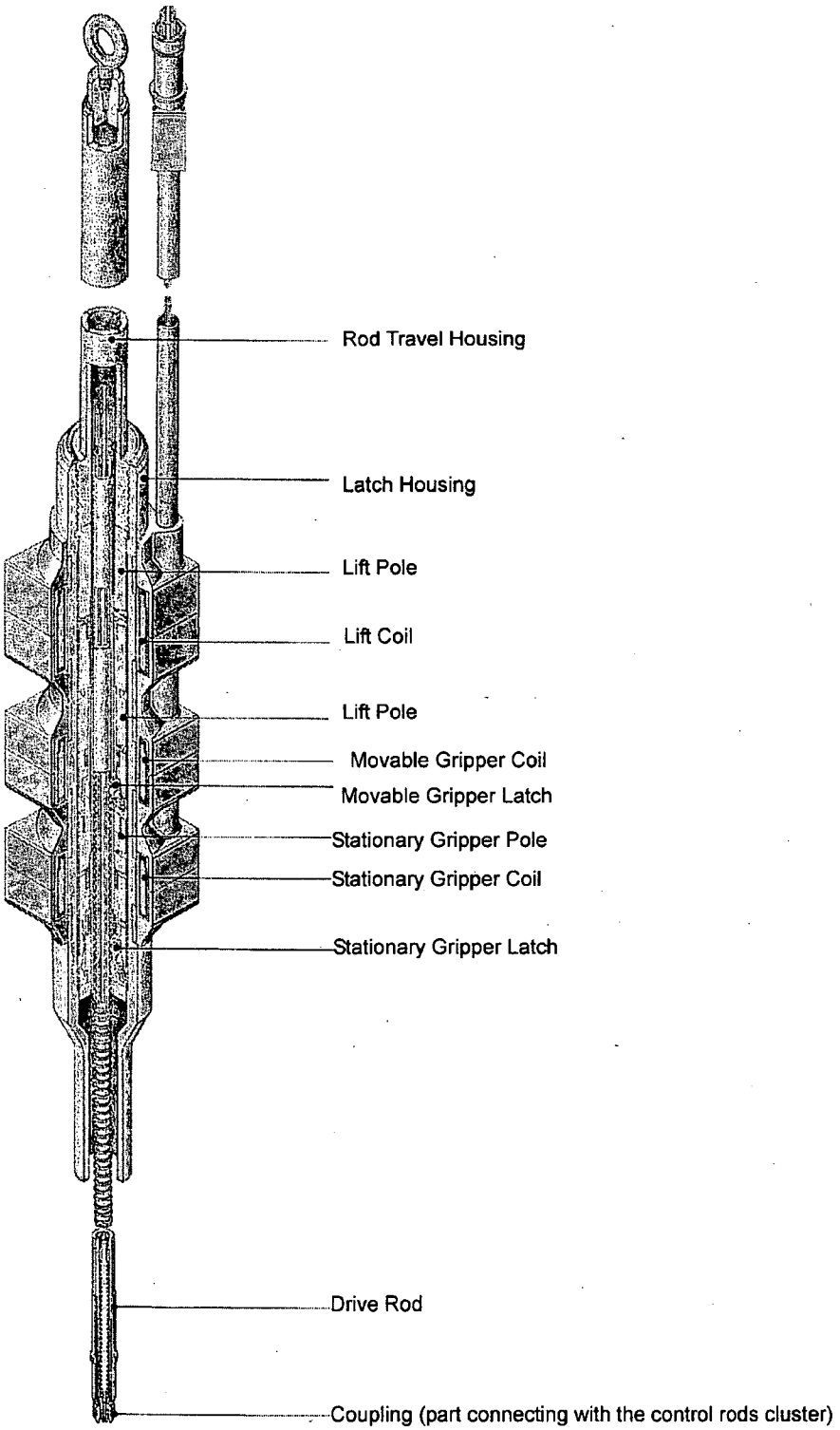


Figure-1 CRDM Cutaway View

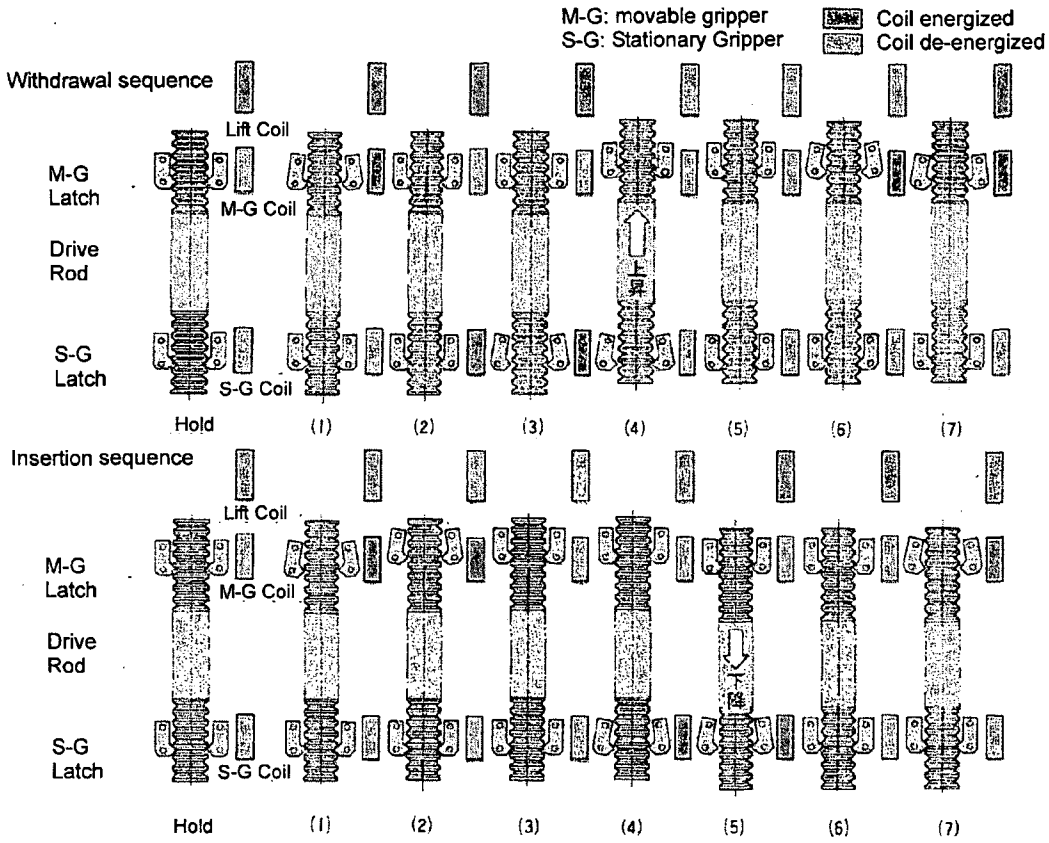


Figure-2 CRDM Operational Sequence

Structural Outline

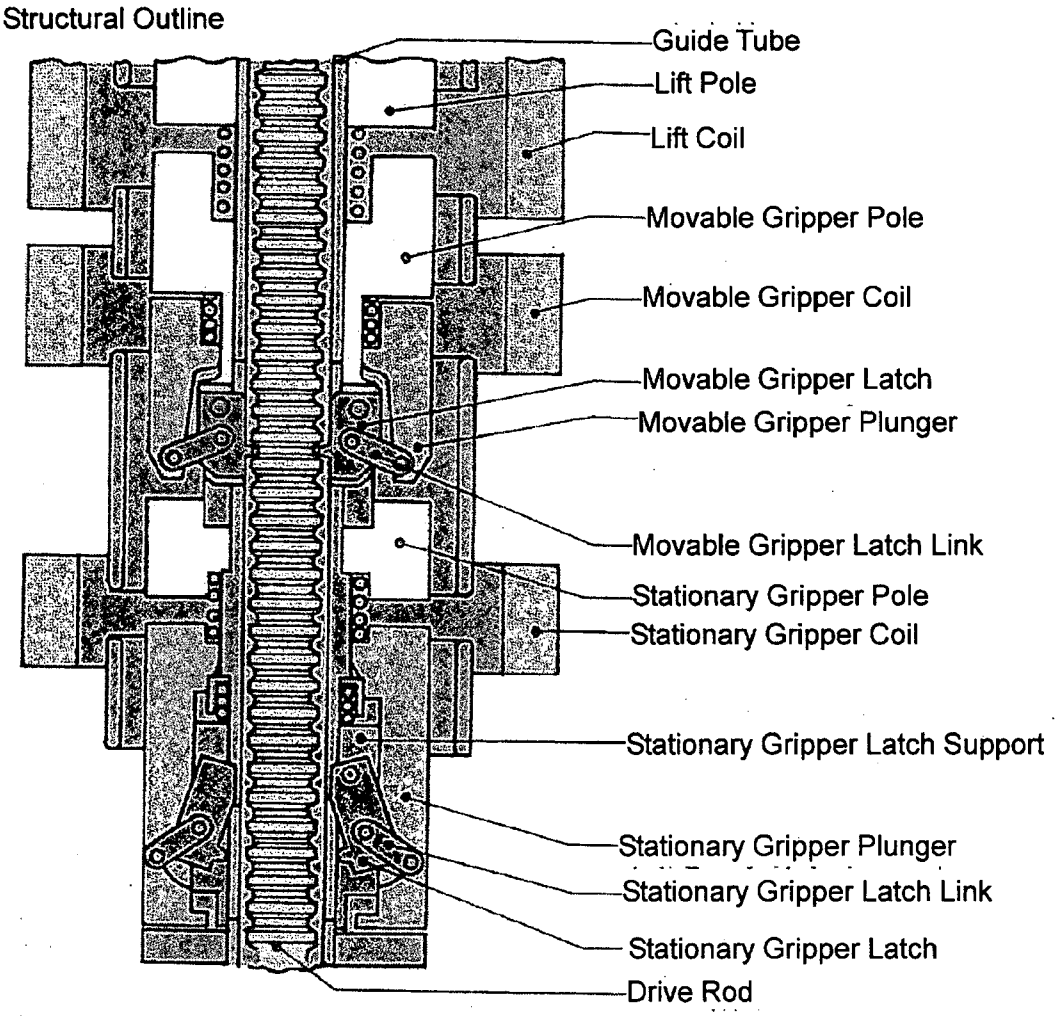
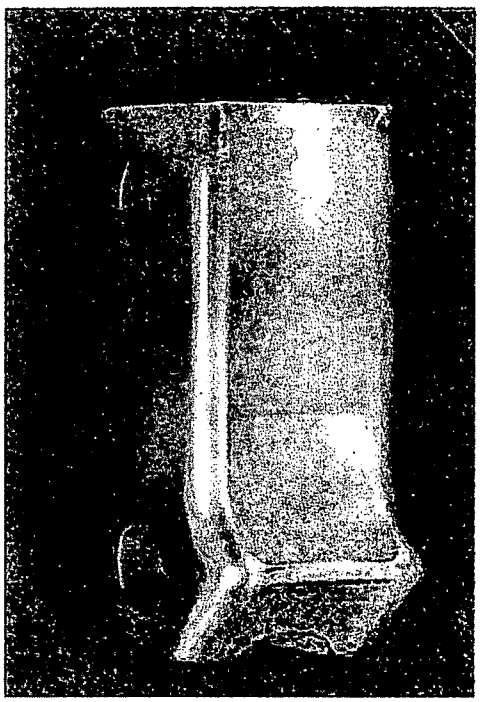


Figure-3 Outline of Latch Assembly

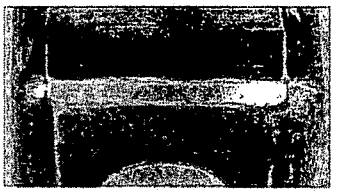
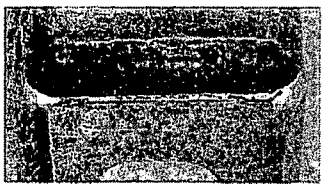
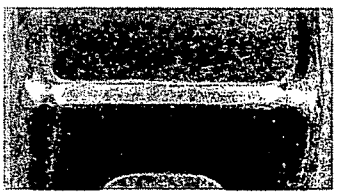
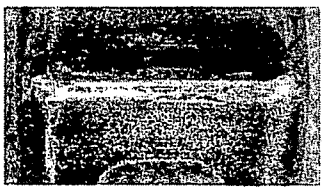
3. Development of the improved latch arm

In 1971 MHI's CRDM went through an endurance test of 3 million steps. The CRDM was designed for 2.5 million operation steps, including load follow operation. The latch assembly was taken apart, and the exact measurements and inspections of its condition before and after the test were compared. The test results showed slight wear on the contact surface of the latch arms with the drive rod's groove, but there was almost negligible wear on the sliding surface on the pins and pinholes, and on the sliding surface by the axial direction movement for withdrawal and insertion the drive rods. However, when the number of operations increased to more than 6 million steps, it was predicted that there would be a slight increase in wear on the latch arm surface. To improve more than two times durability compared with the conventional CRDMs for increasing diversification of plant operation in the nuclear power reactors, it is necessary to improve the latch arm's wear resistance. In conventional latch arm design, cobalt alloy, satellite, cladding is applied onto a base area made of austenite stainless steel for wear resistance of the tip, the pin and the pinholes. To improve wear resistance, we investigated various kinds of surface treatment materials and cladding materials, and we conducted wear resistance test, and resistance test for stress corrosion cracking in the simulated reactor coolant water. As a result of these tests, we found that a latch surface treated with chromium carbide (hereafter Cr_3C_2) thermal spray on the Stellite cladding showed excellent wear resistance and did not have any negative effects to wear of the drive rod engaged with the latches. We conducted long-term wear tests in the simulated reactor cooling water using test sample, which test results showed that there were no adverse effects such as blistering or peeling on the thermal sprayed coating. It was confirmed that Cr3C2 thermal spray coating on the satellite cladding can be used long period. The results of a mock-up wear test which performed in ambient temperature are shown below as sample of the various basic tests we conducted. This test used the drive rods and latch arms that were the same as those using in the actual plants, and it produced wear on the latch tip by dropping the drive rod on the latch arm repeatedly. Photograph 1 shows the improved latch arm, and Figure-4 shows concept of the test equipment. Figure-5 shows comparison the results of mock-up wear test between the improved latch arm and the conventional latch arm under the same conditions. As shown in Figure-5, the improved latch had outstanding wear resistance. Photograph 2 shows the condition of both the improved latch and the conventional latch after six million steps. The conventional latch arm has wear on the tip and its thickness has been reduced. On the other hand, the improved latch arm has a little wear as negligible.



Photograph-1 Appearance of Improved Latch Arm

Condition of the latch arm

	Before Test	After Test (After 6 million Steps)
Current Design Stellite No. 6 Cladding		
Improved Design Stellite No. 6 Cladding + Chromium carbide Thermal spray		

Photograph-2 Appearance of Latch Arm after 6 Million Step Wear Simulation -up Test

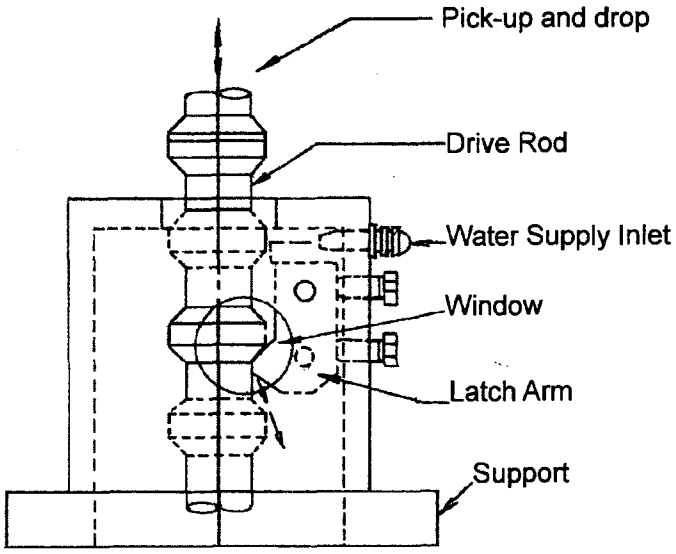


Figure-4 Mock-up for Latch Arm Wear Test

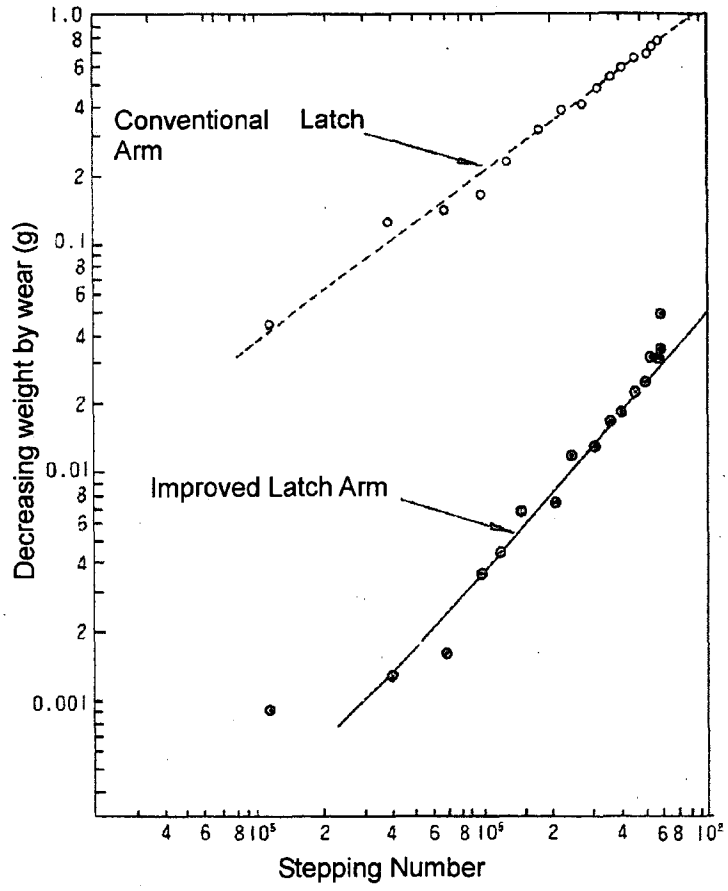


Figure-5 Test Result of Latch Arm Wear Mock-up Test

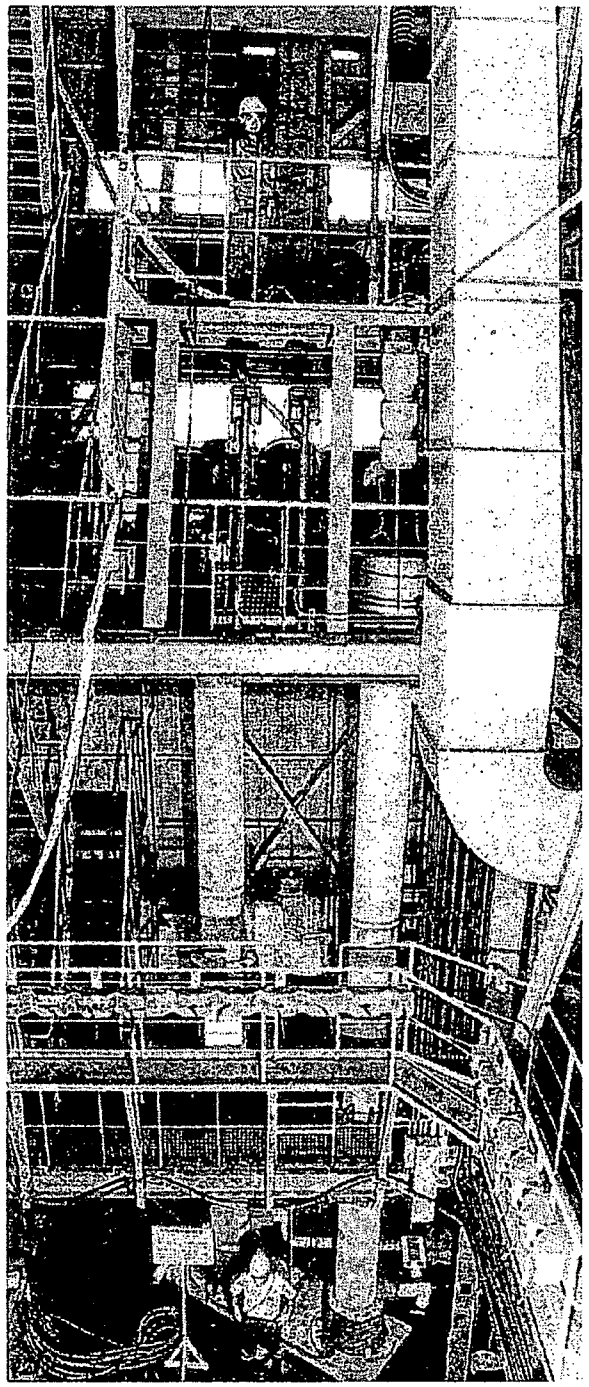
4. Outline of endurance test

Photograph 3 shows the test vessel, Photograph 4 shows the set-up conditions, and Photograph 5 shows the control cabinet. The test vessel is connected to the pressurizer and the heater, and pressure and temperature are controlled to simulate actual plant condition by the control cabinet. We produced a CRDM with improved latch arms and carried out the endurance test under the conditions shown in Table 2. To confirm durability more than six million steps, the endurance test was continued until reaching ten million steps. During the endurance test, five times of the intermediate inspections were performed. As for the intermediate inspection, decrease the test equipment temperature to ambient temperature, and performed view test on the latch arm tips to confirm wear condition by the fiberscope inserted into the latch assembly. At every 50,000 steps during the test, we checked current trace of the operating coils and sounds by latch mechanism action using the electromagnetic oscilloscope to confirm operability that maintain stepping function at rate speed, maximum speed of 114 cm per minute (45 in / min).

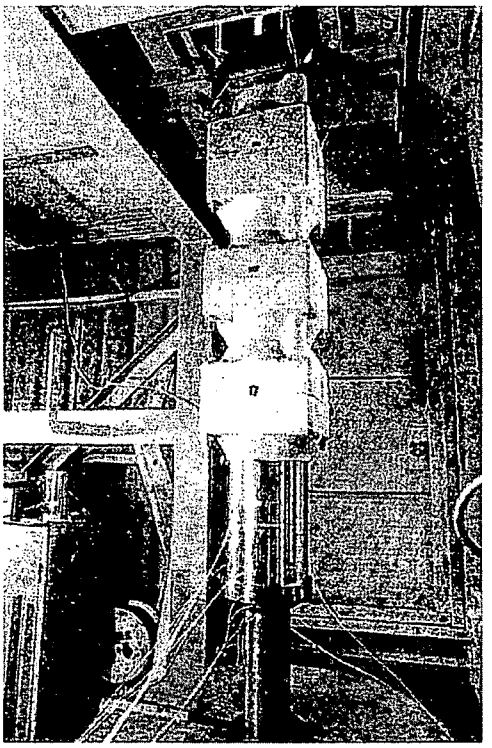
Table 2 CRDM Endurance Test Conditions

Model of CRDM	L-106A
Latch arm	Cr_3C_2 thermal spray
Drive line weight	147 kg (*) (324 lbs)
Test temperature	From 280 degrees C to 290 degrees C (From 536 degrees F to 554 degrees F)
Test pressure	157kg/cm^2 (2250 psia)
Environment	In pure water

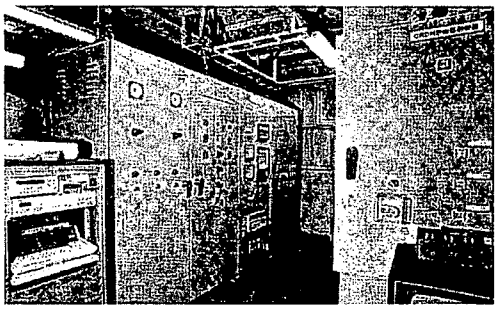
Note(*) Drive line weight is consisted the weight of the drive rod for standard 4 loop and control rods, and margin considering drug force in the actual plant condition.



Photograph-3 CRDM Test Vessel



Photograph-4
Set Up View of CRDM Assembly on Test Vessel
(Part of View for Coil Stuck Assembly)



Photograph-5
Control Cabinet of CRDM and Test Facility

5. Results of the endurance test

Even after ten million steps, the CRDM with the improved latch arms can be operated normally at rated speed without any mechanical or electrical problems. Below we show the results obtained.

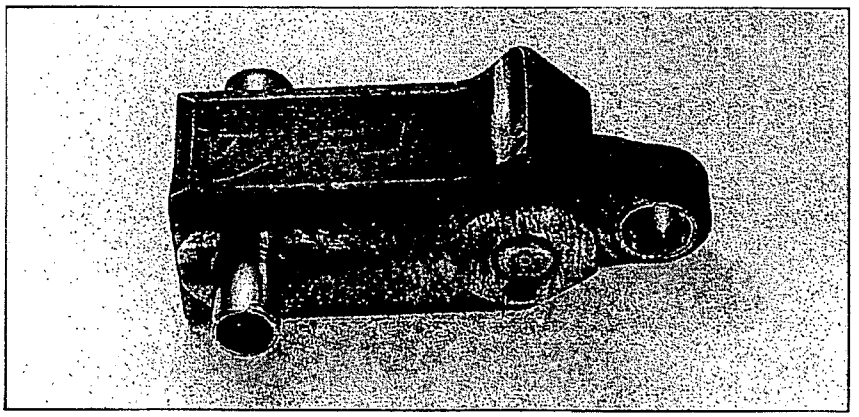
□ After six million steps the chromium carbide thermal spray coating had almost disappeared due to wear and peeling, but the Stellite, which is base material of the tip of the latch arm, had almost no wear. After ten million steps wear on the tip had developed, and its thickness had become small. After the test, we measured the amount of wear on the latch tip in the disassembling inspection. Most wear latch arm in the set, six latch arms per set, to the tested CRDM that had been worn down to knife edge condition at the tip. However there was no adverse effect on the CRDM's operational function. Photograph 6 shows an external view of the latch arms after the test.

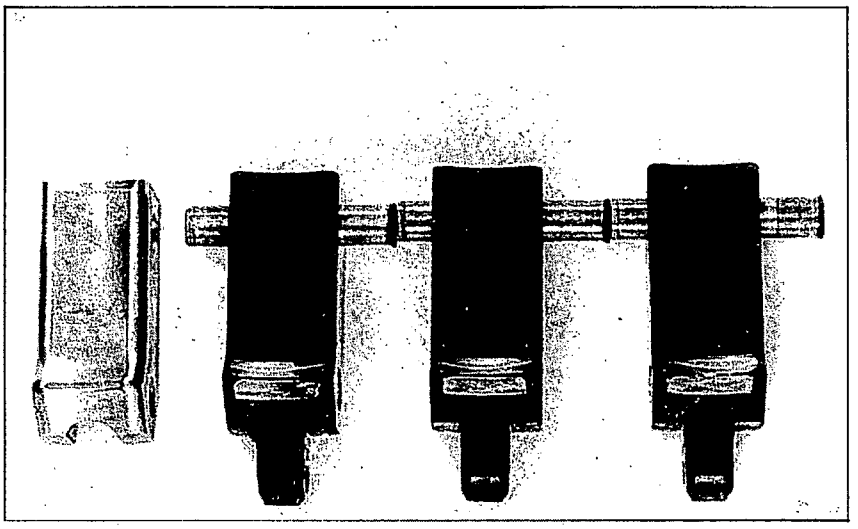
□ There was little wear on the pins and pinholes. At the largest wear area, diameter of the pin and the pin holes are increase or decrease about 0.5 mm (0.02 in).

□ Parts that slid in the axial direction such as the magnetic poles and the plungers showed partial wear on the chrome plating but the amount of wear was about 0.1 mm (0.004 in).

□ In the test, the number of passing times at the latch portion per one groove of the drive rod was 60,000 excursions, which is 120,000 steps. The wear of the grooves of the drive rod was within a permissible range.

□ There was not any un-permissible damage on the all parts such as fatigue. We confirmed that improved CRDM has enough durability such as wear resistance and structural integrity.



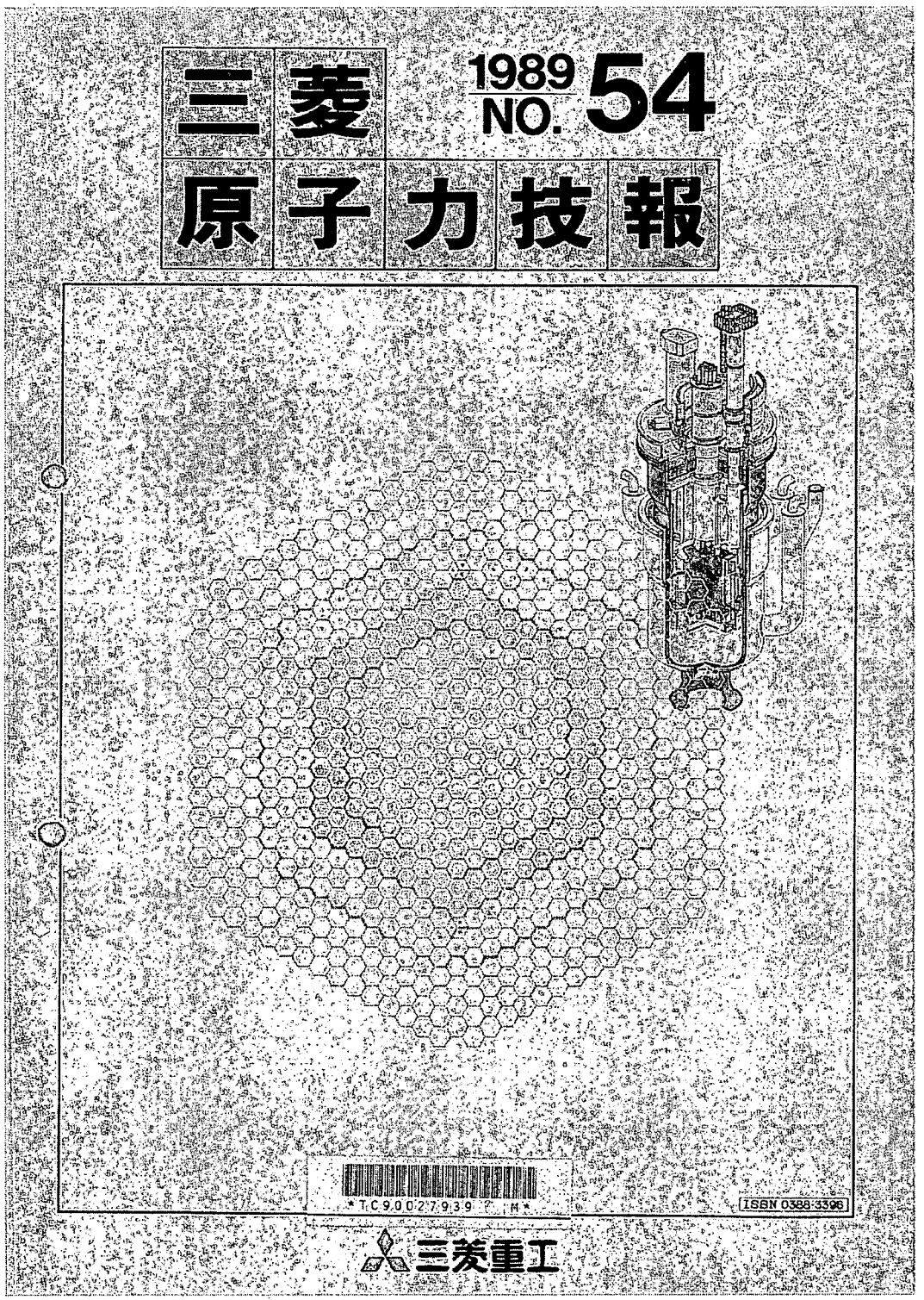


Photograph-6 Appearance of Latch Components After Endurance Test
(After 10 Million Steps of Operation)

6. Afterword

As for the plant operation become more diverse in a nuclear plant, improvement in the durability of CRDMs is requested. Mitsubishi Heavy Industries has developed an improved latch arm with a surface treatment of chromium carbide thermal spray on the tip, and we have shown the durability of improved CRDM in the ten million step test. The test was conducted in high-pressure and high-temperature water conditions which conditions were simulated actual plant condition. We are planning to apply the CRDM with improved latch arms to improve life cycle and reliability of the CRDM for plants which required many operation steps on CRDMs which are under construction or that will be constructed.

Yoshinori Takata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PWR制御棒駆動装置の耐久性向上
Improvement of CRDM Durability for PWR Plant

1. まえがき

L－106A型制御棒駆動装置（Control Rod Drive Mechanism 以下 CRDM という）は原子炉容器の蓋上部に設置し，駆動軸下部のカップリングを介して原子炉内の制御棒クラスタ（以下制御棒という）と接続して，原子炉の出力を制御するために制御棒を引き抜き，挿入あるいは保持する装置である。通常時は駆動速度を毎分10cmから114cmでの範囲で細かい調整ができる。一方，原子炉の緊急停止時には，駆動電流を切ることにより制御棒が自重で炉心内に落下・挿入され，原子炉の核反応を停止する方向に働くので，いわゆるフェール・セーフ化された高い安全性を有している。

最近のプラントでは，周波数制御運転，高性能の負荷追従運転など運転方式の多様化が望まれており，CRDMの耐久性向上が求められている。当社はラッチアーム（以下ラッチという）の爪先にクロムカーバイド溶射の表面処理を施した改良型ラッチを開発し，実機プラントの運転環境を模擬した高温高圧水中条件下で1000万ステップの耐久試験を実施して，運転多様化に対応した設計条件である600万ステップ以上の使用に耐えることを確認した。

2. CRDM設計の概要

三菱標準型CRDMの断面図を図－1に，また設備仕様の概要を表－1に示す。CRDMは，以下の4つの部分組立品から構成されている。制御棒は，3種類の作動コイルがシーケンシャルに励磁・非励磁状態になることにより，図－2のように引抜きまたは挿入される方式である。この方式は完全に密封された圧力容器の外部から電磁力により内部のメカニズムを駆動するので，圧力バウンダリに貫通部を設ける必要がなく，圧力バウンダリの信頼性を損なうことはない。

ラッチアセンブリは駆動軸を保持して，駆動軸をステップ状に引抜き・挿入するメカニズムである。メカニズムの概略を図－3に示す。ラッチアセンブリは3組の磁極・プランジャと2組のラッチ部分（円周方向に1組当り3個）で構成されており，図－2の一連の動作により駆動軸を約16mmずつ上下に移動する。ラッチアセンブリは圧力ハウジングに内蔵されており，作動部分は運転中常に冷却材で潤滑されている。

圧力ハウジングはラッチハウジング，駆動軸ハウジングなどからなり，CRDMの圧力バウンダリを構成している。ラッチハウジングはラッチアセンブリを内蔵し外部には，コイルアセンブリを支持している。駆動軸ハウジングは駆動軸を引き抜いた時，駆動軸を収納する。

駆動軸は外径約45mmの管状で，外面にはラッチと噛み合うように約16mmのピッチで溝を設けている。駆動軸の下端には制御棒を接続するためのカップリングを設け，内部には制御棒を連結・切り離しするための取りはずし軸を組み込んでいる。CRDMを取付けた原子炉容器の蓋を一体で開放すると，駆動軸は炉心に残り，取りはずし軸を操作することにより駆動軸と制御棒の連結・切り離しを行う。

コイルアセンブリは，ラッチハウジングの外周にはめ込んで据え付ける構造になっている。コイルアセンブリには，ラッチアセンブリに対応して3組の作動コイルを内蔵しており，制御電流によりシーケンシャルに励磁または非励磁状態になる。

図-1 CRDM 断面
CRDM Cutaway View

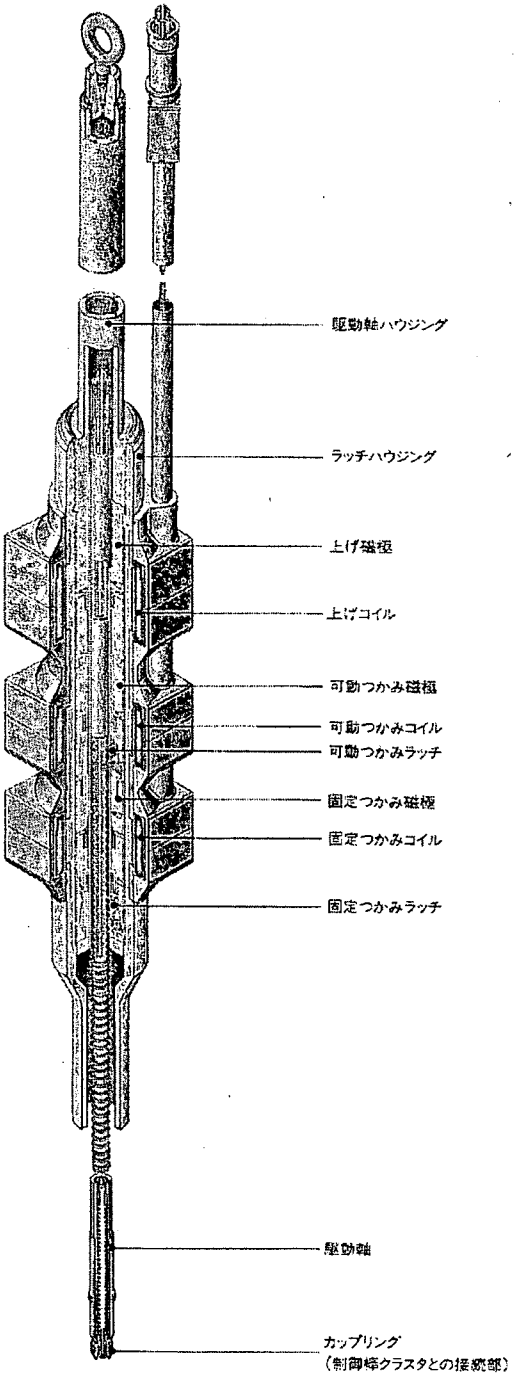


表-1 CRDM 設備仕様
CRDM Equipmental Specification

数 量 (出力制御用必要数)	
2ループ	29
3ループ	48
4ループ	53
駆動方式	
通常運転時	ラッチ式磁気ジャック駆動
トリップ時	重力による落下
駆動速度	
通常運転時挿入・引抜速度	約10~114cm/min
トリップ時挿入時間 (全ストローク85%挿入)	約2秒
設計圧力	175kg/cm²g
設計温度	343℃

図-2 CRDM の作動シーケンス
CRDM Operational Sequence

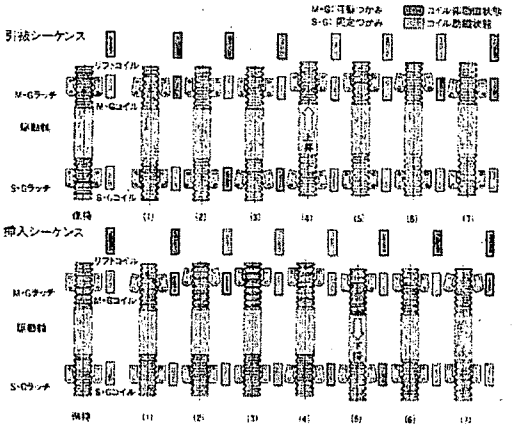
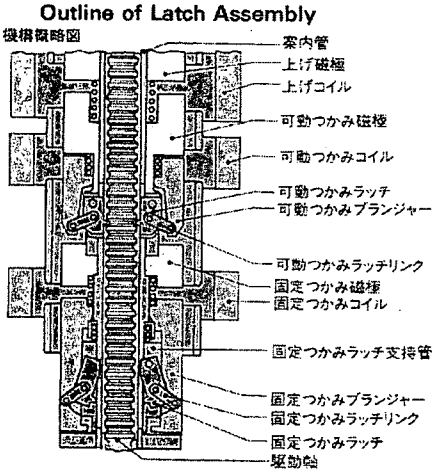


図-3 ラッチアセンブリ概略
Outline of Latch Assembly



3. 改良型ラッチアームの開発

当社のCRDMは1971年に，負荷追従運転を含む作動回数の設計値250万ステップに対して300万ステップの耐久試験を実施した。試験後にラッチアセンブリを分解して，試験前後の寸法検査を行った結果，ラッチの駆動軸の溝山と噛み合う当り面は軽微な摩耗があったが，ピンやピン穴および軸方向にスライドする面の摩耗はほとんどなかった。しかし作動回数が約600万ステップ以上になるとラッチアーム当り面の摩耗が若干増加すると予想された。原子力発電所の運転多様化に対応してCRDMの耐久性を従来の2倍以上に向上するには，ラッチの耐摩耗性を向上する必要がある。従来型ラッチはオーステナイト系ステンレス鋼の母材にステライト合金を肉盛りし，爪先およびピンとピン穴部の耐摩耗性の向上を図っている。これをさらに改良するため各種の肉盛り硬化材と表面処理材について，摩耗模擬試験や原子炉内の冷却材を模擬した水中での応力腐食割れ試験を実施した。これらの試験の結果，現用盛金材であるステライトにクロムカーバイト（以下Cr_3C_2という）溶射の表面処理を施したラッチが優れた耐摩耗性を有し，ラッチと噛み合う駆動軸の摩耗に悪影響がないことを確認した。また，原子炉内の冷却材を模擬した水中で，試験片による長時間の摩耗試験を実施した結果，溶射層に剥離やふくれなどの異常がなく，長期の使用に耐えうることを確認した。各種の基礎試験結果のうち，常温で実施したモックアップ摩耗試験の結果を以下に示す。この試験は実機と同じラッチと駆動軸を用い，軸を落下させてラッチ爪先に摩耗を生じさせる試験である。改良型ラッチを写真－1に，試験装置の概念を図－4に示す。従来型ラッチと改良型ラッチを同一の条件で試験した結果，図－5に示すように改良型ラッチは優れた耐摩耗性を示した。600万ステップ後の摩耗状況を写真－2に示す。従来型ラッチは爪先の摩耗が進展して厚みが小さくなっているが，改良型ラッチはほとんど摩耗していない。

写真-1 改良型ラッチアームの外観
Appearance of Improved Latch Arm

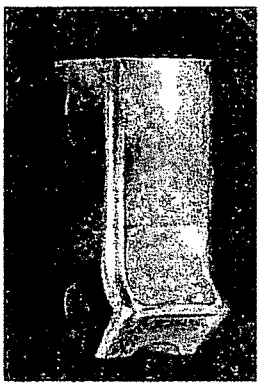


写真-2 摩耗模擬試験（600万ステップ）後のラッチアーム外観
Appearance of Latch Arms After Wear Mock-up Test (After 6 Million Steps)

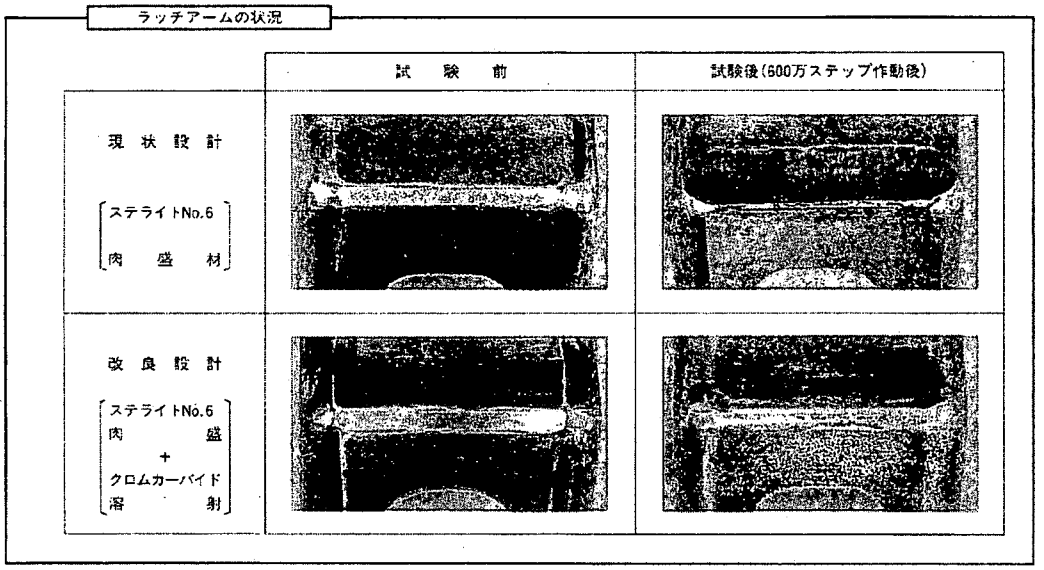


図-4 ラッチアーム摩耗模擬試験装置
Mock-up for Latch Arm Wear Test

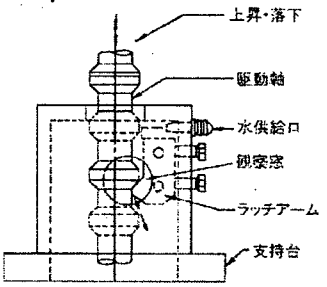
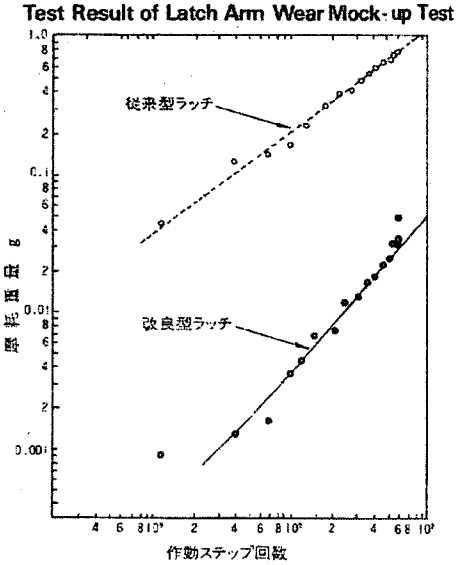


図-5 ラッチアーム摩耗模擬試験結果
Test Result of Latch Arm Wear Mock-up Test



4. 耐久試験の概要

試験容器を写真－3に，CRDM設置状態を写真－4に，また制御盤を写真－5に示す。高温高圧の試験容器には加圧器と加熱器が接続され，制御盤により試験容器の温度と圧力を実機プラントと同様の条件に保持する。改良型ラッチを組み込んだCRDMを製作し表－2に示す条件で耐久試験を実施した。設計条件の600万ステップ以上の耐久性を確認するため5回の中間検査を行いながら1000万ステップまで試験を継続した。中間検査は試験装置の温度を常温まで下げ，ラッチアセンブリの内部にファイバスコープを挿入して，ラッチ爪先の摩耗状況を観察した。また，試験中5万ステップごとに電磁オシログラフにより作動コイルの電流と作動音を監視し，定格速度（最大毎分114cm）で正常に作動していることを確認した。

写真-3 CRDM試験容器
CRDM Test Vessel

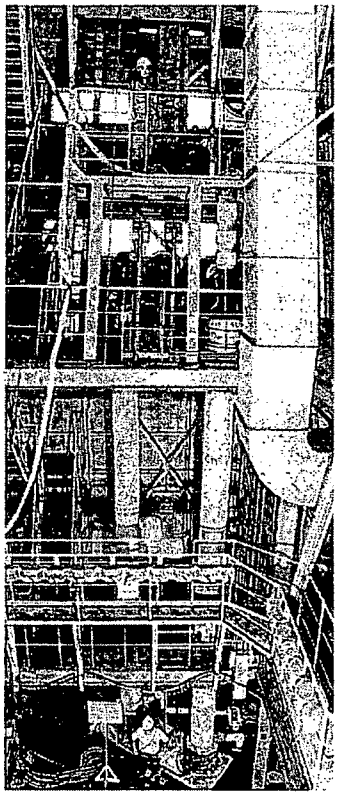


写真-4 CRDM取付け状態（コイルアセンブリ部）
Set Up View of CRDM Assembly on Test Vessel
(Part of View for Coil Stuck Assembly)

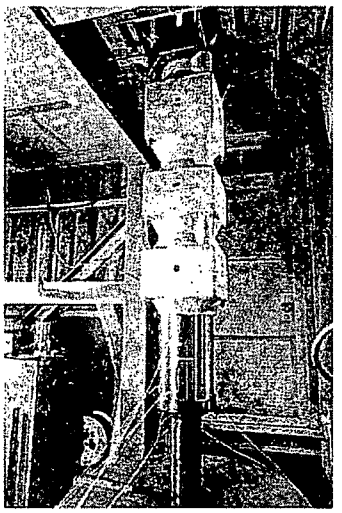


写真-5 CRDM・試験設備の制御盤
Control Cabinet of CRDM and Test Facility

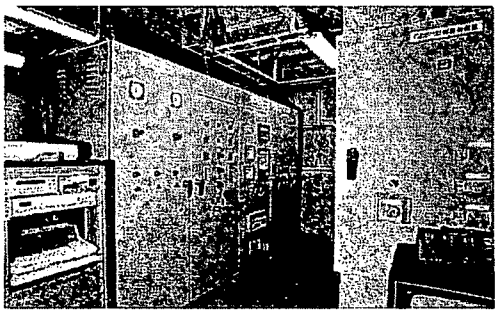
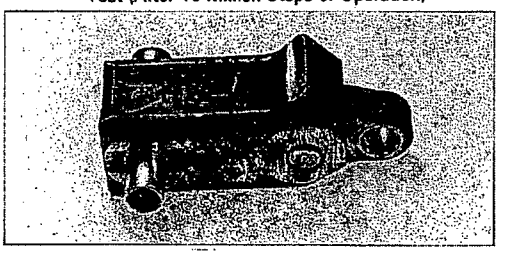
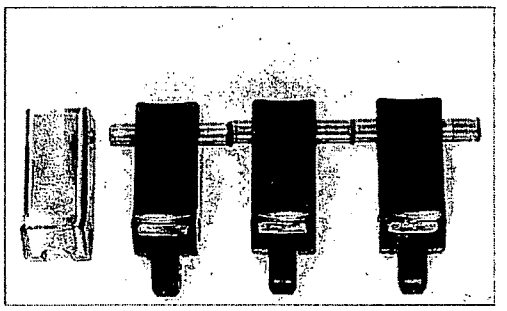


表-2 CRDM耐久試験条件
CRDM Endurance Test Condition

試験条件	
(1)制御棒駆動装置；標準L-106A型	(4)試験温度；280～290℃
(2)ラッチアーム；Cr_3C_2溶射	(5)試験圧力；157kg/cm²
(3)駆動系重量；147kg(*)	(6)試験環境；純水
注記(*) 標準4ループプラントの駆動軸と制御棒の重量に実機プラントでの摩擦抗力を加算した値。	

写真-6 耐久試験(1000万ステップ)後のラッチ部外観
Appearance of Latch Components After Endurance Test (After 10 Million Steps of Operation)





5. 耐久試験の結果

改良型ラッチを組み込んだCRDMは1000万ステップにおいても機械的・電気的になんら問題なく，定格速度で正常に作動した。以下に得られた結果を示す。

①駆動軸と当たるラッチのCr_3C_2溶射層は，ほぼ600万ステップで摩耗と剥離によりなくなったが，母材のステライトはほとんど摩耗していなかった。1000万ステップでは，爪先の摩耗が進展し厚みが小さくなった。試験後，分解検査を行いラッチ爪先の摩耗量を測定した。6個のラッチの中で最も摩耗量の大きいラッチでは，爪先がほぼナイフエッジ状に摩耗したが，CRDMの作動特性にはなんら影響がなかった。試験後のラッチ部外観を写真－6に示す。

②ピンとピン穴部の摩耗は少なく最大の箇所でも直径で0.5mm程度であった。

③磁極・プランジャなどの軸方向にスライドする部品ではクロムメッキが部分的に摩耗していたが摩耗量は0.1mm程度であった。

④試験により経験した駆動軸の1山当りのラッチ通過回数は，6万往復回であったが，摩耗は許容できる範囲であった。

⑤すべての部品は疲労破壊などの損傷がなく，CRDMは摩耗並びに，強度的にも高い耐久性を有することが確認できた。

6. あとがき

原子力発電プラントの運転方式が多様化するにつれて，CRDMの耐久性向上が求められている。当社はラッチの爪先にCr_3C_2溶射の表面処理を施した改良型ラッチを開発し，実機プラントの運転環境を模擬した高温高圧水中条件下で1000万ステップの耐久性を確認した。建設中および今後建設されるプラントでは，作動回数の多いCRDMに改良型ラッチを適用し，CRDMの長寿命化と信頼性の向上を図る計画である。(高田　良則)